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Introduction to R&D of APWR at SWCR

906B0020A Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 10 No 5, Oct 89 pp 4-8 [MS received 25 May 89]

[Article by Sun Yufa [1327 3768 4099] and Min Yuanyou [7036 0337 0147] of the Southwest Center for Reactor Engineering Research and Design, Chengdu: "A Brief Introduction to R&D of the APWR at the Southwest Center for Reactor Engineering Research and Design"]

[Text] **Abstract**

This article provides a brief introduction to tracking and R&D work by the Southwest Center for Reactor Engineering Research and Design (SWCR) concerning advanced pressurized-water reactors. It describes the design goals of a Chinese-style advanced pressurized-water reactor (APWR) power plant (AC-600), characteristics of the preliminary program, and tentative ideas for future work. The AC-600 has the advantages of small engineering investments, good safety, high reliability, short construction schedules, and a very good ability to interconnect with pressurized-water reactor (PWR) power plants China has already constructed or is preparing to build. It is the main direction in China's development of nuclear power plants.

Key words: advanced pressurized-water reactor, AC-600, spectral shift control, gadolinium burnable poison, passive safety system, system simplification.

I. Introduction

Energy resources are a serious problem which affect China's four modernizations drive. China's electric power requirements and the growth of each type of energy resource based on surveys by energy experts in the relevant departments are shown in Tables 1 and 2.

Table 1. Forecast of Domestic Power Needs 1985-2030

Year	1985	1990	2000	2023
Power needs ($x10^8$kWh)	4,037	5,500	12,000	51,860

Table 2. Forecast of Growth of Various Types of Power in China 1980-2000

	1980		1990		2000	
	10^8kWh	Percent	10^8kWh	Percent	10^8kWh	Percent
Coal-fired	1,981	62.9	4,550	75.8	8,900	74.2
Oil-fired	533	17.7	2,520	4.2	200	1.7
Hydropower	582	19.2	1,200	20.0	2,600	21.6
Nuclear	—	—	—	—	300*	2.5
Total	3,006	100	6,000	100	12,000	100

*Nuclear-power total installed capacity $=5.5 \times 10^6$kW

By the year 2000, coal-fired power must reach 890 billion kWh, which will require about 500 million tons of raw coal each year. Data from state projections, however, show that it will be very hard for the state to provide more than 400 million tons of coal for power generation, so it will be very hard for coal-fired power to exceed 720 billion kWh. Moreover, there are enormous problems in building coal, oil, and other conventional power plants in areas such as transport, reducing environmental pollution, and so on, and hydropower stations are subject to regional and seasonal restrictions. For this reason, development and utilization of nuclear power and accelerated development of nuclear power are essential. Thus, it has been projected that there will be more extensive development of nuclear power in the 21st century and that there will be a substantial increase in nuclear power as a proportion of China's overall energy-resource structure. China's nuclear power industry has just begun to develop. The Qinshan first-stage-project 300 MW_e nuclear power plant and Daya Bay 2 x 900 MW_e nuclear power plants are now under construction. Several more 600 MW_e nuclear power plants will be built to meet our electric power requirements in the year 2000. If we wish to accelerate development of our nuclear power industry, we must formulate nuclear power plants, technical lines, and technical policies on the basis of trends in the development of nuclear power technologies in foreign countries and China's own national conditions.

II. Nuclear-Power Technology Development Trends in Foreign Countries and China's Countermeasures

The accident at the Three Mile Island Nuclear Power Plant had definite effects on the development of nuclear power in the world, which even stopped for a time. However, because of the substantial increase in demand for energy and growing shortages of energy resources, nuclear power will continue to develop as a clean, safe, and economical energy resource after 1990 and into the next century. Forecasts based on surveys by the IAEA indicate that the world's total installed nuclear-power generating capacity will reach 480 to 600 GW_e in the year 2000, and nuclear power will become a primary energy resource in the 21st century. Excellent economy, safety, and reliability are the main goals in developing nuclear power. Because light water reactors (LWR's), especially PWR's, are the most mature reactor types at the present time, rich operating experience has been

accumulated. For the past 10 years, many nations which have nuclear power plants (such as the United States, Japan, Sweden, the Soviet Union, West Germany, France, etc.) have been working hard to improve LWR's to make them safer and more reliable while maintaining their economic advantages. They have developed advanced pressurized-water reactors (APWR), advanced boiling water reactors (ABWR), process inherent ultimately safe reactors (PIUS), and other types of LWR technologies, and they have made notable progress. In France, because the government decided to process all the fuel and had large amounts of plutonium on hand, they developed the RCVS (Reacteur Convertible a Variation de Spectre) on the basis of nuclear-power-plant standardization and moved to spectral-shift reactors to take full advantage of their plutonium and to use spectral-shift control to reduce the cost of fuel cycles and increase load-following capabilities. In the United States, on the basis of several decades of experience with PWR power plants, the Westinghouse Corporation proposed a program for an advanced passive inherently safe PWR power plant (AP-600). The AP-600 provides inherent safety in a nuclear power plant and its cost—30 percent less than that for existing 600 MW_e nuclear power plants—is only $1,270 per kW. Reports from Westinghouse indicate that AP-600 technologies are maturing and they expect to begin construction in 1994 and to reach criticality in late 1997. Definite progress also has been made in other nations, such as Sweden's PIUS reactor and Japan's APWR and ABWR. Related articles in this issue of HE DONGLI GONGCHENG provide information on the ALWR development situation in various nations of the world.

When China develops nuclear power, we should consider these national conditions:

1. China is a developing nation and lacks a firm economic foundation. We cannot simultaneously develop several reactor types, but should make one type of reactor the primary reactor type and strive to reduce the relative investments in a nuclear power plant to increase its economic benefits.

2. China has already built several nuclear reactors. We are now building a 900 MW_e and a 300 MW_e PWR nuclear power plant, and we are preparing to build several MW_e PWR nuclear power plants. Thus, nuclear power development in China should utilize the designs for these PWR's and our experience in building and operating them to increase the proportion of domestic production.

3. China has already imported a complete set of 600 MW_e conventional island steam-turbine technologies.

4. China has a dense population, so when building a nuclear power plant we must consider safety and reliability as well as simplification of emergency-response-plan implementation after a major accident.

5. The principle for developing nuclear power in China is "With China as the main factor, Chinese-foreign cooperation." In today's world, PWR nuclear power plants continue to be a safe, reliable, and economical energy resource accounting for 54 percent of nuclear power plants. This is especially true in development of advanced reactors, where foreign countries have made substantial progress. Thus, we should consider international cooperation to reduce development funds and time, and we do not have to build model reactors.

On the basis of development trends with APWR technologies in foreign countries and China's national conditions, we feel that we should manufacture 600 MW_e standard PWR's as a foundation for actively developing international cooperation and for developing a Chinese-style 600 MW_e passive-safety APWR nuclear power plant to catch up with advanced world levels as quickly as possible.

III. A Brief Introduction to China's R&D Work on APWR's

To track world developments in the nuclear power industry and search out a direction for nuclear power development which conforms to China's national conditions, the [China National] Nuclear Industry Corporation (CNNIC) considered technical forces and experimental facilities at SWCR. They assigned the task of developing an APWR to SWCR and established the APWR Development Group for this purpose in late 1986. After nearly 3 years, we have completed these main items of work:

1. We surveyed and studied the development situation and trends for advanced LWR's in foreign countries and compiled "Current Situation and Progress Trends for Nuclear Power in the World," "Current Situation in Advanced Light Water Reactor Development," "High-Conversion Light-Water Reactors," and other survey research reports to prepare technical data for SWCR to undertake APWR development work.

2. On the basis of survey research, we provided CNNIC with proposals and preliminary program ideas for developing a 600 MW_e APWR in China. The proposals took full advantage of China's experiences in building reactors and PWR nuclear power plants and integrated with China's nuclear power development plans to develop APWR's as quickly as possible and to make a shift from existing 600 MW_e nuclear power plants to APWR nuclear power plants the main direction in nuclear power construction in China. Our proposal was approved by CNNIC.

3. We completed a preliminary outline design and did preliminary calculations and safety analysis for the nuclear-island portion of China's 600 MW_e APWR nuclear power plant (called the AC-600) and suggested related scientific research projects.

4. Under arrangements by CNNIC, SWCR sent experts on several occasions to international APWR symposia held by the IAEA and relevant state organs. We also

prepared several reports on technical questions concerning the AC-600 and drew a warm reaction from symposium participants. We were entrusted by the IAEA with holding an international APWR symposium directed by SWCR [here] at Chengdu, which has laid a foundation for SWCR to develop international cooperation on APWR's.

5. In accordance with a request by CNNIC, we made additional improvements in the outline design for the AC-600 and included the AC-600 development plan in the Eighth 5-Year Plan (1991-1995) for the national economy, and pushed to have it made a key state scientific research topic during the Eighth 5-Year Plan.

IV. Primary Design Goals and Characteristics of the AC-600

1. Primary Design Goals

The primary design goals for the AC-600 are: 1) Increase reactor safety and reliability. 2) Improve economy. 3) Raise power-plant utilization ratios. 4) Reduce construction schedules and extend power-plant lifespans. On the basis of these requirements, the primary design indices for the AC-600 are:

Construction cost	About 20 percent lower than existing 600 MW_e PWR nuclear power plants
Core-meltdown probability	$< 1.5 \times 10^{-6}$/reactor-year
Power-plant utilization ratio	> 85 percent
Fuel-cycle length	Greater than or equal to 18 months
Construction period	About 4 years
Power-plant lifespan	60 years
Job radiation for personnel	0.5 to 1.0 person-Sv [person-Sieverts]/year
Natural-cycle power	15 to 30 percent of rated flow rate

2. Primary Design Characteristics

The thermal power of the AC-600 APWR is 1,820 MW, the electric power is 600 MW, the total height of the reactor is 19.1 meters, the maximum outer diameter is 5.04 meters, the total coolant flow rate is 31,200 tons/hour, the number of coolant-system loops is two, the reactor inlet temperature is 287.7°C, the outlet temperature is 324.5°C, and the working pressure is 15.5 MPa. The core is composed of 137 19x19-type fuel assemblies.

To increase the economy, safety, and reliability of the AC-600 and increase the power-plant's utilization ratio, reduce construction periods, and extend lifespan, we mainly adopted measures in three areas: an advanced core, a passive safety system, and system simplification in the outline design. The characteristics are:

(1) Advanced Core

(i) Mechanical spectral-shift-control technologies are employed. A certain number of compressed water rods are installed in the fuel assemblies to change the core water-uranium volume ratio to improve fuel utilization, reduce fuel-cycle costs, and improve reactor safety.

(ii) Addition of gray control rods and application of gadolinium burnable poison. Besides black control rods, gray control rods are also added, and the gadolinium burnable poison Gd_2O_3 is added to the core to increase load-following capabilities. This helps disperse the core power distribution and fully utilize the fuel, and it can substantially reduce the critical boron concentration.

(iii) Addition of a stainless-steel assembly reflecting layer. Stainless-steel assemblies are added in a radial direction in the area surrounding the AC-600 core to form a radial neutron-reflecting layer, reduce neutron leakage, and help reduce the irradiation fast flux of the thermal screen and pressure vessel and extend the useful life of the pressure vessel.

(iv) Reduction of core power density. The dimensions of the fuel assemblies were increased in the AC-600 core design to reduce the core power density and increase the safety margin. The AC-600 core is 366 cm high, the core equivalent diameter is 317 cm, the linear power density is 12.3 kW_t/m, and the mean volume power density is 63.02 MW_t/m^3.

(2) Passive Safety Systems

Safety systems that do not depend on external machinery or electric power, but rely instead on natural laws and materials characteristics to perform their function are called passive safety systems. In the design of the safety systems for the AC-600, except for retaining active equipment for low-pressure safety injection pumps to perform long-term recycling during a loss of coolant accident, all other systems are passive safety systems. The main ones are:

(i) Emergency core residual heat-removal system. The emergency core residual heat-removal system is composed of an emergency water supply tank, an emergency air cooler, and the corresponding pipes, valves, and control instruments. There are two sets for the two loops, and the two sets constitute the AC-600 emergency core residual-heat removal system.

(ii) Safety injection system. The AC-600 safety injection system is similar to the safety injection system in existing PWR nuclear power plants. It is divided into high-, moderate-, and low-pressure injection systems and the corresponding recirculation system. The high- and moderate-pressure injection portions are composed mainly of two core water-compensation tanks having the same pressure as the reactor coolant. The moderate-pressure injection portion is composed mainly of two safety injection tanks with working pressures of about 5.2 MPa. Both the high- and moderate-pressure injection systems are passive.

(iii) Containment-vessel cooling system. This system uses entirely passive equipment and is composed mainly

of a containment-vessel-cooling water storage tank, cooling water pump, and so on at the top of the containment vessel. It replaces the safety spray system in existing PWR nuclear power plants and does not require equipment cooling water or other intermediate-cycle cooling medium, which saves on investment and improves the operational reliability of the system.

These passive safety systems ensure the performance of emergency reactor-decay heat removal, control of the amount of core feedwater, short-term loss-of-coolant accident (LOCA) injection, long-term LOCA recirculation, containment-vessel spraying, containment-vessel cooling, and other safety functions after transients and accidents at nuclear power plants.

(3) System Simplification

(i) The reactor coolant system (RCS) is composed of two loops. Each loop includes a steam generator and two shielded pumps linked together in a single unit with the steam generator. This simplifies the primary equipment support and eliminates a supporting system required by the shaft-seal main pumps, the main-pump shaft seal-water system.

(ii) The RCS employs a dense configuration with curved pipes having a large radius of curvature. This greatly reduces on-site welding and shortens construction periods.

(iii) Linking the main pumps and steam generators together in one unit eliminates the U-shaped transitional segment of the main pumps. This improves safety after a LOCA, reduces the length of the main pipes, lowers resistance in the primary system, and improves the natural circulation capabilities of the system.

(iv) Adoption of a passive safety system and reduction of the boron concentration in the reactor coolant eliminates or simplifies the auxiliary feedwater system, residual heat-removal system, chemical and volume-control systems, boron recovery system, and high-pressure safety injection pumps. Initial statistics indicate a 70-percent reduction in the number of valves in a single-loop auxiliary system, about a 50-percent reduction in instruments, and about an 80-percent reduction in the nuclear second- and third-stage pumps.

(v) Safety-stage [i.e., containment-area] equipment is installed completely within the containment shell, thus reducing safety-stage building area and saving on expense.

Besides these three main points, our design also employs an advanced control room, modular construction, and other advanced technologies in order to improve the performance of the AC-600, reduce construction periods, and lower construction costs.

V. Preliminary Ideas for Future Work

1. In the previously compiled Eighth 5-Year Plan based on China's national conditions, push to have the AC-600 included among key scientific research topics during the Eighth 5-Year Plan.

2. On the basis of mutual benefit, actively develop international exchange and international cooperation to immediately grasp experiences from all nations of the world in the area of developing APWR's and further improve the AC-600 program.

3. Complete independent preliminary feasibility research on the AC-600 before 1990, including plant site selection, technical programs and measures, preliminary safety and environmental analysis, models for foreign cooperation, analysis of domestic production, economic progress, and scheduling.

4. When conditions permit, make good preparations for key experimental projects to enable experimental research work to begin after the project is established.

In summary, we feel that making a rapid transition from existing 600 MW_e PWR nuclear power plants to APWR nuclear power plants should be the primary direction for nuclear power development in China. If the project can be included among state projects to attack key scientific research topics during the Eighth 5-Year Plan, China will be able to build a Chinese-style AC-600 nuclear power plant during the Ninth 5-Year Plan (1996-2000). This will move China's nuclear power industry into the ranks of the world's advanced nuclear technologies.

Core Design

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[Article by Zhang Zongyao [4545 1350 5069], Li Dongsheng [2621 0392 3932], Luo Wei [7482 0251], Zhang Senru [1728 2773 1172], Fu Shouxin [0102 1343 0207], and Wu Lin [0702 3829] of the Southwest Center for Reactor Engineering Research and Design, Chengdu: "AC-600 Core Design"]

[Text] **Abstract**

This article describes the nuclear core design, thermal hydrodynamics, and shielding design of the AC-600. To improve fuel utilization and increase reactor safety, the core design adopts a low-power-density core, a rather large negative-reactivity temperature coefficient, mechanical spectral-shift control, gadolinium burnable poison, gray rods, stainless-steel reflecting layers, and other advanced technologies.

Key words: spectral-shift control, low-power-density core, gadolinium burnable poison, stainless-steel reflecting layer, gray rod.

I. Introduction

The design and research goals for the AC-600 are increased safety and reliability of nuclear power plants, and lower construction costs and fuel consumption. Improving the design of the reactor core is an extremely important aspect of attaining these goals. For this

reason, this article provides a concise introduction of the AC-600 nuclear core design, thermal hydrodynamics, and shielding design.

II. Reactor Core Design

1. Design Criteria

(1) Under any working conditions, to assure that no departure from nucleate boiling (DNB) or core meltdown occurs, the maximum power non-uniformity coefficient cannot exceed design limits.

(2) From hot-state zero power to full power in the reactor, the moderator temperature coefficient (MTC) should be a negative value.

(3) Rod-blocking [i.e., rod-insertion-limit] criteria are met, meaning that under any working conditions, when the most efficient control rod is blocked at [i.e., withdrawn from] the top of the core, the other control rods can still shut down the hot reactor and the hot-reactor-shutdown margin is no less than 1.5 percent. Using two independently configured control systems can enable the reactor to attain and sustain cold-reactor-shutdown working conditions.

(4) When a fuel equilibrium cycle is attained, the average proportional burnup for the removed fuel assemblies attains the predicted value. The maximum proportional burnup for the fuel assemblies must not exceed the design value to ensure the mechanical integrity of the fuel elements.

2. Outline of Reactor Core Design

(1) Computing model and program. The main programs used for reactor core design were the FAN8-B_1 approximate fast-neutron energy-spectrum computation program, the CLTHBC grid-element thermal-spectrum and burnup computation program, and the CMSS two-dimensional coarse-grid spread method fuel-management computation program. To test the reliability of the reactor core design, we did several international reference problems and nuclear power-plant PWR nuclear core-design checking computations with these programs and the nuclear data. The results of the analysis showed that the precision of the calculations of core effective-multiplication coefficient, power distribution, fuel-cycle length, heavy-isotope output, and other important performance parameters derived using these computing programs attained IAEA-suggested indices and could meet the requirements of engineering design.

(2) Core configuration. The AC-600 core is composed of 137 square fuel assemblies of the 19 x 19 type. Each fuel assembly has 296 fuel rods, one measurement tube, and 16 guide tubes [guide "thimbles"] for insertable control rods or compressed-water rods. Each guide tube occupies the position of 2 x 2 fuel rods. The core-configuration and fuel-assembly cross sections are shown in Figures 1 and 2, respectively.

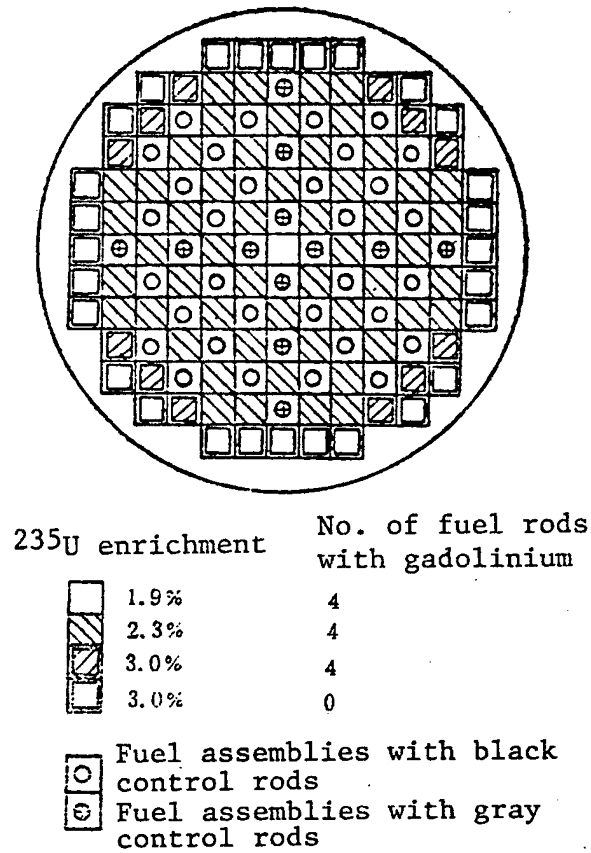

Figure 1. Configuration of AC-600 Core

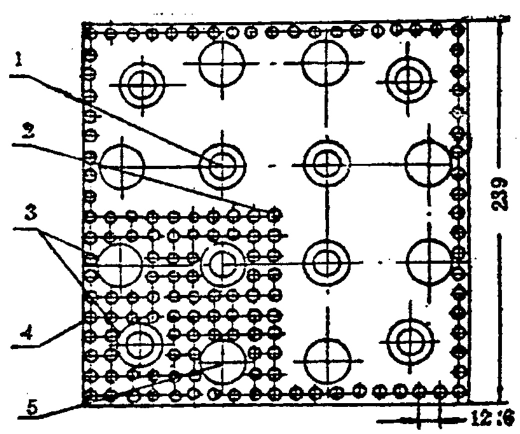

Figure 2. Schematic Diagram of AC-600 Fuel Assembly.
Key: 1. Control rod 2. Measurement tube 3. Guide tube 4. Fuel rod 5. Compressed-water rod

The ^{235}U enrichments in the UO_2 fuel loaded into the first-cycle core are, respectively, 1.9 percent (45 assemblies), 2.3 percent (48 assemblies), and 3.0 percent (44 assemblies). The fuel assemblies with the maximum

enrichment are placed around the outside of the core. The fuel assemblies of the two remaining enrichments are placed in a checkboard configuration within the core. The equilibrium refueling enrichment is 3.4 percent and the fuel-cycle length is 18 months.

The first-cycle core uses Gd_2O_3 mixed evenly with natural uranium as a burnable poison. The percentage content by weight of the Gd_2O_3 is 3.0 percent. Four gadolinium-bearing fuel rods are placed in each of the fuel assemblies with a ^{235}U enrichment of 1.9 percent and 2.3 percent. Gadolinium-bearing fuel rods are placed in only 12 of the modules with a ^{235}U enrichment of 3.0 percent (see Figure 1). Subsequently, each new refueling assembly placed into each cycle also contains gadolinium-bearing fuel rods.

Reactivity control in the core is achieved using a solid control-rod system and boron system (chemical and volume control systems). The core is configured with 52 control-rod clusters; rods are divided into two types, black rods and gray rods. Each black control-rod cluster is composed of eight Cd-In-Ag rods; the black rods are used mainly for normal and accidental reactor shutdown. The gray control-rod clusters, each composed of four Cd-In-Ag rods and four stainless-steel rods, are used mainly for load following.

A mechanical spectral-shift system is installed in the core. Sixteen compressed-water rods are inserted into the fuel assemblies which do not have control rods inserted. Eight compressed-water rods are inserted into the modules which do have control rods. They are linked to the compressed-water-rod hydraulic-drive mechanism in the fuel assemblies which do not have control rods inserted. The compressed-water rods occupy the position of 2 x 2 fuel rods. The material is Zr-4.

A reflecting-layer assembly composed of stainless-steel rods is installed in the core reflecting layer. It is used to reduce the irradiation flux of the core thermal shield and pressure vessel.

The main parameters of the AC-600 reactor core design are listed under Scheme I in Table 1.

Table 1. Schedule of APWR and PWR Core Design Parameters

Parameter	Scheme I (AC-600)	Scheme II (APWR)	Scheme III (PWR-SWCR)*
Rated thermal power, MW	1820	1820	1820
Net electric power, MW	600	600	600
Reactor operating pressure, MPa	15.5	15.5	15.5
Core height, cm	366	366	366
Core equivalent diameter, cm	317	292	267
Average power density, W/cm^3	63.02	74.19	88.91
Coolant inlet temperature, °C	287.7	288.17	287.01
outlet temperature, °C	324.5	325.75	324.75
average temperature, °C	306.10	306.10	306.9
Average temperature of fuel rods, °C	637.03	608.40	609.65
Number of fuel assemblies	137	145	121
dimensions, cm x cm	23.9 x 23.9	21.4 x 21.4	21.4 x 21.4
Fuel-rod arrangement	19 x 19	17 x 17	17 x 17
Fuel-rod grid spacing, cm	1.26	1.26	1.26
core block diameter, cm	0.805	0.805	0.819
casing outer diameter, cm	0.95	0.95	0.95
casing thickness, cm	0.064	0.064	0.057
Number of fuel rods per assembly	296	264	265
First-cycle ^{235}U enrichment, percent	1.9/2.3/3.0	2.0/2.4/3.1	1.9/2.3/3.0
Equilibrium refueling ^{235}U enrichment, percent	3.4	3.4	3.1
First-cycle ^{235}U initial loading, kg	1627.1	1632.0	1343.2
UO_2 initial loading, t	68.518	65.370	56.064
Burnable poison material	Gd_2O_3-NU	Gd_2O_3-NU	boron glass
First-cycle ^{155}Gd initial loading, kg	2.9736	2.6518	—
^{157}Gd initial loading, kg	3.2064	2.8594	—

Table 1. Schedule of APWR and PWR Core Design Parameters (Continued)

Parameter	Scheme I (AC-600)	Scheme II (APWR)	Scheme III (PWR-SWCR)*
Percentage by weight of Gd_2O_3 in Gd-bearing fuel rods, percent	3.0	3.0	—
^{10}B linear density, g/m	—	—	0.623
Number of burnable-poison rods in first cycle	436	384	704
Number of control-rod clusters	52	53	33
Number of control rods per cluster	8	24	24
Control-rod material	Cd-In-Ag/S.S.	Cd-In-Ag/S.S.	Cd-In-Ag
Compressed-water-rod material	Zr-4	—	—
Total number of compressed-water rods	1776	—	—
Number of compressed-water rods per assembly	8/16	—	—
Core water-uranium volume ratio	2.042/2.369**	2.009	1.9336
First-cycle fuel-cycle length, EFPD***	722	557	>350
Unit ^{235}U loading cycle time, EFPD/kg	0.444	0.341	—

*The PWR-SWCR is a 600 MW_e PWR nuclear power-plant design developed at SWCR, Chengdu.

**Values when the compressed-water rods are inserted/withdrawn.

***Equivalent/effective full-power days.

(3) Fuel-management computation and analysis. At 70 percent of the AC-600 core cycle time (about 400 EFPD [equivalent/effective full-power days]), burnup-analysis calculations were made for the operating pattern with all the compressed-water rods pulled out simultaneously. Figures 3 and 4 show, respectively, the variation curves for the core effective [neutron] multiplication factor k_{eff} and critical boron concentration C_B for the first-cycle period. Figures 5 through 8 show the core normalized power distribution [K_{xy}] and neutron-flux distribution for different fuel periods in the first-cycle period. Because a mechanical spectral-shift control system was employed, the core k_{eff} is increased about 3.8 percent Δk at 400 EFPD, which is equivalent to a net increase of about 100 EFPD in the fuel period.

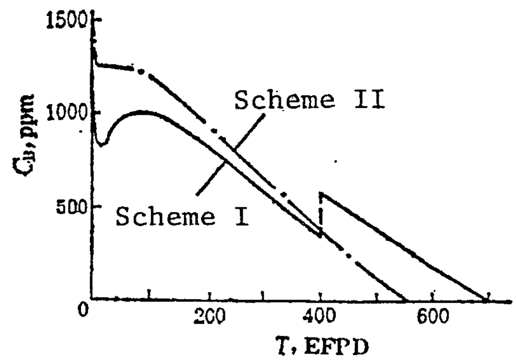

Figure 4. Curve of Critical Boron Concentration C_B for AC-600 Core During First Cycle

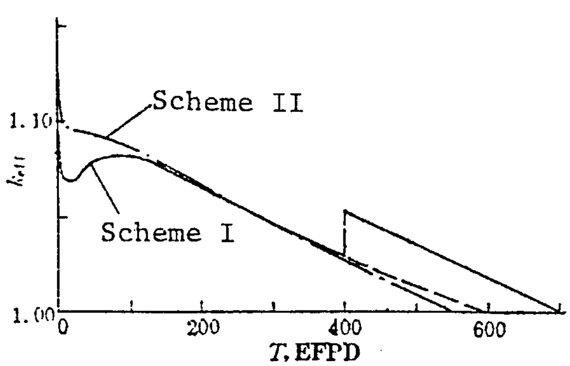

Figure 3. Curve of k_{eff} for AC-600 Core During First Cycle

K_{xy} —
Fast neutron flux, $10^{14}n/cm^2 \cdot s$ —
Thermal neutron flux, $10^{14}n/cm^2 \cdot s$ —

				1.1888 2.1569 0.4123	0.7129 1.2273 0.2463	
			1.1119 2.1636 0.4927	1.0496 2.1471 0.5516	0.9643 1.6966 0.3369	
		1.0631 2.0867 0.4710	1.0310 2.0963 0.5425	1.1193 2.2063 0.4955	1.0340 2.0729 0.4833	0.7207 1.2196 0.2501
	0.9941 1.9431 0.4407	0.9829 2.0010 0.5170	1.0831 2.1246 0.4800	1.0505 2.1365 0.5525	1.1241 2.2309 0.4969	0.9727 1.7124 0.3343
0.8066 1.7218 0.4256	0.9055 1.8348 0.4767	1.0362 2.0390 0.4589	1.0137 2.0576 0.5335	1.1116 2.1881 0.4922	1.0612 2.1567 0.5584	1.0174 1.7793 0.3502

Figure 5. K_{xy} and Neutron-Flux Distribution at 0 EFPD During First Cycle in the AC-600 Core

K_{xy} —
Fast neutron flux, $10^{14}n/cm^2 \cdot s$ —
Thermal neutron flux, $10^{14}n/cm^2 \cdot s$ —

				1.0302 1.9248 0.3500	0.5884 1.0428 0.2002	
			1.1395 2.2856 0.4650	1.0101 2.0769 0.5026	0.8371 1.5138 0.2877	
		1.2258 2.4601 0.5218	1.1357 2.3252 0.5658	1.1080 2.2290 0.4711	0.9750 1.8801 0.4165	0.5746 1.0007 0.1966
	1.2468 2.4944 0.5316	1.1958 2.4508 0.5956	1.2079 2.4207 0.5144	1.0854 2.2251 0.5403	1.0031 2.0285 0.4256	0.7466 1.3655 0.2613
1.0820 2.3205 0.5374	1.1782 2.4035 0.5885	1.2524 2.5164 0.5329	1.1636 2.3792 0.5799	1.1366 2.2849 0.4833	0.9884 1.9816 0.4825	0.7760 1.4116 0.2617

Figure 6. K_{xy} and Neutron-Flux Distribution at 100 EFPD During First Cycle in the AC-600 Core

K_{xy} —
Fast neutron flux, $10^{14}n/cm^2 \cdot s$ —
Thermal neutron flux, $10^{14}n/cm^2 \cdot s$ —

				1.3028 2.1987 0.4715	0.8604 1.3336 0.3024	
			1.1125 2.0209 0.4980	1.0748 2.0934 0.5489	1.0784 1.7350 0.3919	
		0.9659 1.7920 0.4413	0.9576 1.8633 0.4920	1.1132 2.0237 0.4958	1.0457 1.9839 0.4734	0.7973 1.2214 0.2819
	0.8034 1.6378 0.4041	0.8678 1.6926 0.4452	0.1006 1.8274 0.4507	0.9913 1.9329 0.5084	1.1450 2.0777 0.5068	1.0128 1.6373 0.3548
0.7920 1.5051 0.3973	0.8063 1.5644 0.4136	0.9362 1.7010 0.4184	0.9080 1.7700 0.4668	1.0749 1.9525 0.4601	1.0416 2.0213 0.5323	1.0349 1.6756 0.3641

Figure 7. K_{xy} and Neutron-Flux Distribution at 400 EFPD During First Cycle in the AC-600 Core

At the end of the first cycle (550 EFPD), the average specific burnup of the fuel assemblies was 14,800 MW x d/t(U) and the maximum burnup could reach 16,500 MW x d/t(U). The computed results for k_{eff} for the first three cycle periods for the AC-600 core are shown in Figure 9.

K_{xy} —
Fast neutron flux, $10^{14}n/cm^2 \cdot s$ —
Thermal neutron flux, $10^{14}n/cm^2 \cdot s$ —

				1.2196 2.1011 0.4813	0.8885 1.3341 0.3220	
			1.0746 1.9769 0.5231	1.0286 2.0228 0.5535	1.0597 1.7094 0.4143	
		1.0213 1.8741 0.4941	0.9639 1.8854 0.5291	1.0768 1.9827 0.5221	1.0177 1.9300 0.4926	0.8294 1.2565 0.3085
	1.0019 1.8310 0.4824	0.9356 1.8291 0.5115	1.0273 1.8860 0.4982	0.9802 1.9205 0.5381	1.0960 2.0162 0.5284	0.9941 1.6154 0.3707
0.9133 1.7417 0.4876	0.9161 1.7796 0.4990	1.0126 1.8559 0.4862	0.9447 1.8470 0.5183	1.0572 1.9429 0.5122	1.0060 1.9657 0.5502	1.0075 1.6415 0.3779

Figure 8. K_{xy} and Neutron-Flux Distribution at 550 EFPD During First Cycle in the AC-600 Core

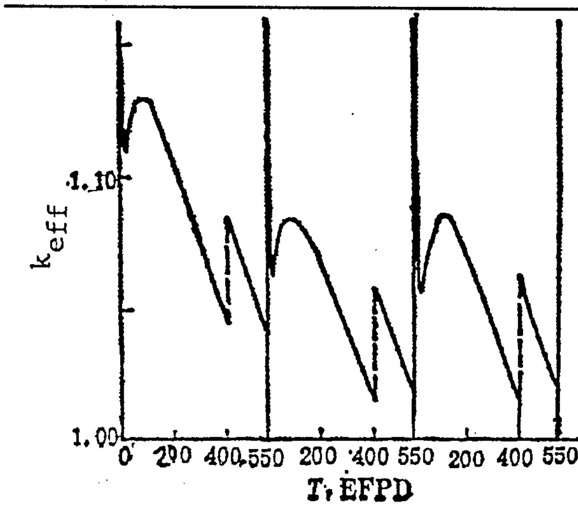

Figure 9. Curve of k_{eff} During First Three Cycles for AC-600

(4) Reactor-shutdown analysis and control computations. To ensure safe operation and safe shutdown of the reactor, the reactor core design must provide adequate reactivity-control capabilities. The solid control rods, gadolinium-bearing burnable poison rods, and soluble-boron control system installed on the AC-600 reactor can assure safe reactor shutdown. Table 2 provides the results of calculations on reactor shutdown.

Table 2. Results of Computations for Reactor Shutdown and Control Capacity

Cycle time	Operating conditions	Burnable poison	Compressed-water rods	Black control rods	Gray control rods	C_B, ppm	k_{eff}
Early	Hot-state zero power	Yes	Yes	Fully inserted	Fully inserted	1117.3*	0.924177
Early	Hot-state zero power	Yes	Yes	1 cluster blocked	Fully inserted	1117.3	0.941671

Table 2. Results of Computations for Reactor Shutdown and Control Capacity (Continued)

Cycle time	Operating conditions	Burnable poison	Compressed-water rods	Black control rods	Gray control rods	C_B, ppm	k_{eff}
Early	Hot-state zero power	Yes	Yes	Fully inserted	1 cluster blocked	1117.3	0.941573
Early	Hot-state zero power	Yes	No	1 cluster blocked	Fully inserted	1117.3	0.958439
Late	Hot-state zero power	No	No	1 cluster blocked	Fully inserted	0.0	0.974811
Early	Cold-state	Yes	Yes	Fully extracted	Fully extracted	2050.0	0.95
Early	Cold-state	Yes	Yes	Fully inserted	Fully inserted	1570.0	0.95

*C_B =1117.3 ppm is the critical boron concentration for hot-state full-power operating condition at 0 EFPD.

The following conclusions can be drawn from the data in Table 2.

a. In the later part of the reactor fuel cycle, under hot-state zero-power conditions, the core has no burnable poison, no soluble boron, and no compressed-water rods. When the control rod with the highest efficiency is blocked at the top of the core and all of the remaining control rods are inserted into the core, the reactor shutdown margin is 2.52 percent, which meets blocked-rod criteria.

b. In the early part of the first fuel cycle in the reactor under cold-state conditions, all the control rods are pulled out of the core to ensure a boron concentration of 2,050 ppm required for cold reactor shutdown (k_{eff} = 0.95). This value provides the highest requirements for the amount of boron additive in the chemical and volume-control system.

3. Improvements in Reactor Core Design

(1) Mechanical spectral-shift control technologies. The AC-600 core employs mechanical spectral-shift control technologies to improve fuel utilization and increase reactor safety. In the early stage of reactor operation, all the compressed-water rods are inserted into the core to discharge part of the core moderator, water. At this time, the core neutron energy spectrum is rather hard and conversion is rather high, which means that more renewable material (such as ^{238}U) is being converted into fission materials. Moreover, because the core water-uranium volume is rather small and total residual reactivity is rather small, the goal of safe control can be attained without requiring as many control materials as in regular PWR cores, and it avoids the loss of excessive numbers of neutrons drifting into the control materials. In the later periods of operation, the compressed-water rods are pulled out of the core. This increases the water-uranium volume ratio, softens the core neutron energy spectrum, and increases the effective [neutron] multiplication coefficient k_{eff}. Simultaneously, it also achieves full utilization of the already produced nuclear fission in the core. These two factors increase the residual reactivity of the core and lengthen the fuel cycle, thereby attaining the goals of reducing fuel-cycle costs and improving the economy of nuclear power plants.

Table 1 compares the design parameters for a core with compressed-water rods (Scheme I, the AC-600), a core without compressed water rods (Scheme II), and the PWR-SWCR nuclear core. The results of theoretical research show that for reactors with fully identical initial loads of UO_2 fuel, a configuration with mechanical spectral shift can lengthen the reactor fuel cycle by 10 percent compared to one without it. Moreover, lifting the compressed-water rods and lifting them in groups can affect the fuel-cycle length to different degrees (see Figure 10).

(2) Utilization of gadolinium burnable poison and reaction control. The AC-600 core uses a new burnable poison, Gd_2O_3, to disperse the core power distribution and improve fuel utilization.

The control capabilities of gadolinium burnable poison are stronger than those of B_4C and boron glass, and there is an extremely small residual amount of gadolinium at the end of the cycle, so the residual absorption component is small, thereby enabling full use of the fuel. Figure 11 shows a plot of the curves of the residual amounts of ^{155}Gd and ^{157}Gd following burnup in the first cycle. Moreover, because the gadolinium-bearing control rods are easy to install, they offer considerable flexibility in dispersing the power distribution within the assemblies.

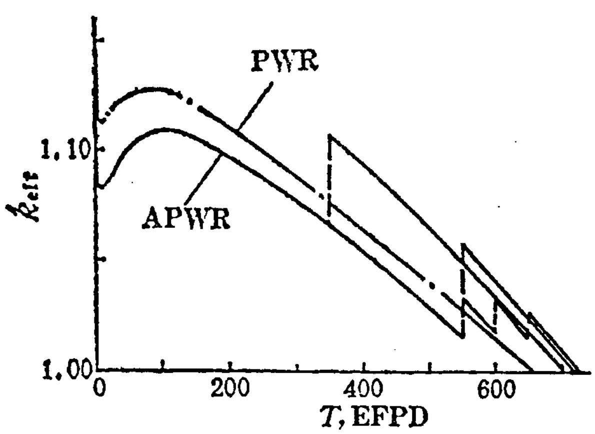

Figure 10. Effects of Pattern in Which Compressed-Water Rods Are Lifted on Fuel-Cycle Length

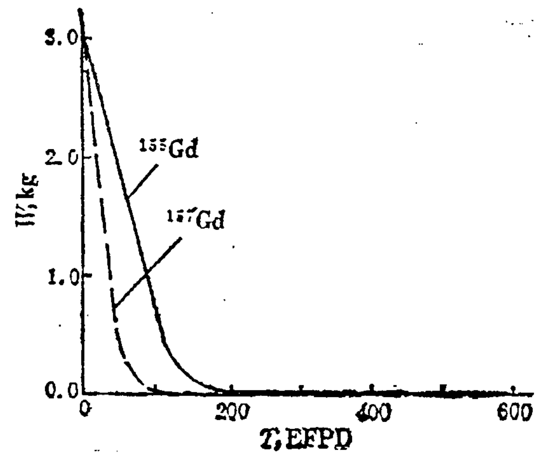

Figure 11. Burnup Curve for Gadolinium Burnable Poison

In the first-cycle period, if the depth of control-rod insertion into the active height of the reactor is varied within a range of 10 to 30 percent (Figure 12), the critical boron concentration C_B is basically held at the 100 ppm level (Figure 13), corresponding to a [change in] reactivity $\Delta\rho$ of 1 percent. With this low critical boron concentration, most excess reactivity is compensated for by the burnable poison Gd_2O_3, which greatly simplifies the boron system.

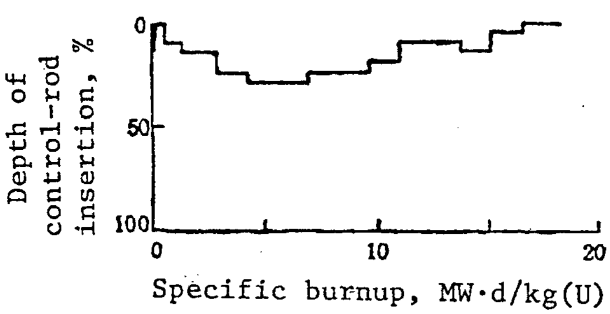

Figure 12. AC-600 Core Control-Rod Operating Patterns

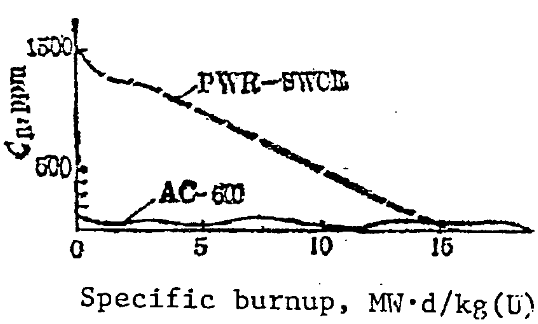

Figure 13. Curve of Critical Boron Concentration in AC-600 Core (During First Cycle)

(3) Stainless-steel-rod assembly reflecting layer. Stainless-steel rod assemblies are installed in the radial exterior region of the AC-600 core to form a radial neutron reflecting layer. This helps reduce the irradiation fast flux of the thermal shield and pressure vessel, extend the useful life of the pressure vessel, and reduce the construction costs of the nuclear power plant.

The results of computations show that after installation of a radial stainless-steel-rod assembly reflecting layer, the pressure-vessel fast-neutron (E > 1 MeV) integral flux is reduced by 60 percent.

(4) Low-power-density core. The design of the AC-600 reactor increases the dimensions of the fuel assemblies to reduce the power density and reduce neutron leakage. The core power density is reduced almost 30 percent compared to that of the PWR-SWCR. This can guarantee that the fuel elements have rather small energy storage and rather low fuel-cladding temperatures in the reactor under rare accident working conditions (category 3 working conditions) and minimum accident conditions (category 4 working conditions), thereby increasing reactor safety.

III. Thermal Hydrodynamics Criteria

1. Design Criteria

(1) Consideration of the effects of negative factors on DNBR [departure from nucleate boiling ratio]. The W-3 formula was used to compute the critical thermal flux density. No departure from nucleate boiling occurs in the heat channel under category 1 or 2 working conditions, and the minimum DNBR must allow a sufficient design surplus to meet the relevant accident working conditions.

(2) Under category 1 and 2 working conditions, the maximum temperature of the fuel in the core must not exceed the melting point of the fuel for the corresponding specific burnup, and the effects of all types of negative factors must be taken into account to allow a sufficient design surplus for category 3 and 4 working conditions.

2. Outline of Thermal-Hydrodynamics Design

The reactor system safety analysis program RSSA was used for analytical computations of the AC-600 reactor thermal hydrodynamics. This program can be used for safety analysis of operating states as well as for steady-state thermal-hydrodynamics computations.

In the calculations, the amount of leakage from the pressure-vessel inlet to the outlet is assumed to be 5 percent of the first-loop total flow component. The initial enthalpy of the coolant at the core inlet is 1,227.3 kJ/kg, the product of the

hot-channel engineering factor and the core-power local-peak factor is 1.17078, and the core-current non-homogeneous-entry-cavity allocation factor is 1.05, so the poorest module entry flow rate is considered to be 5 percent less than the nominal flow rate. The results of the main calculations are shown in Scheme I in Table 3.

Table 3. Comparison of Results of Thermodynamic Calculations for Different Schemes

Parameter	Scheme I (AC-600)	Scheme II (APWR)	Scheme III (PWR-SWCR)
Reactor thermal power, MW_t	1820	1820	1820
Reactor operating pressure, MPa	15.5	15.5	15.5
Reactor inlet temperature, °C	288.18	288.17	187.01
Reactor outlet temperature, °C	325.75	325.75	324.75
Average coolant temperature, °C	306.10	306.10	306.9
Average temperature of fuel elements, °C	637.03	608.40	669.65
Core-coolant mass flow rate, $kg/m^2/h$	$0.7761x10^7$	$0.9086x10^7$	$1.0241x10^7$
Hot-channel-coolant mass flow rate, $kg/m^2/h$	$0.7265x10^7$	$0.8020x10^7$	$0.9416x10^7$
Hot-channel outlet steam content, percent	1.5209	6.745	3.588
Maximum temperature at center of hot-channel elements, °C	1440.0	1489.9	1858.4
Minimum burnout ratio	2.0022	1.7424	1.6501
Average linear power density, kW/m	12.2625	12.9903	15.5669

3. Thermal-Hydrodynamics Design and Parameter Selection

To debate the thermal hydrodynamics design, a thermal-hydrodynamical analysis was carried out for the three core schemes in Table 1.

In Scheme I, the core was composed of 137 fuel assemblies arranged in 19 x 19 rows and columns, for a total of 40,552 elements. Compressed-water rods were installed in 104 assemblies in the core. This is the AC-600 core scheme. The core in Scheme II is composed of 145 assemblies arranged in 17 x 17 rows and columns, for a total of 38,280 elements. There are no compressed-water rods in the core. Scheme III is a PWR-SWCR core composed of 121 assemblies with the fuel elements arranged in 17 x 17 rows and columns. This scheme was the core for a 600 MW_e PWR nuclear power plant designed by SWCR in 1980.

Table 3 lists the results of thermal hydrodynamics computations for each of the cores described above. It is apparent from Table 3 that the main thermal hydrodynamic parameters in Scheme I and Scheme II, such as minimum burnout ratio, highest temperature in the center of hot-channel elements, element linear power density, and so on, are all safer than the corresponding parameters for the PWR-SWCR. Scheme I is the AC-600. The design is more rational and safer, its minimum burnout ratio is larger, the hot-channel outlet steam content is lower, and the hot-channel element center temperature is also lower than for Scheme II.

Figure 14 plots the minimum burnout ratio ($DNBR_{min}$) for these core design schemes with changes in reactor inlet temperature. Figure 15 plots the minimum burnout ratio ($DNBR_{min}$) for each core scheme under different reactor operating pressures. Figure 16 compares the hot-channel outlet steam content for each of the core schemes under different reactor operating pressures.

The results of these computations lead to this conclusion: the AC-600 core (Scheme I) with a compressed-water rod-control system can satisfy thermal hydrodynamics design criteria and has a greater margin of safety than the other programs.

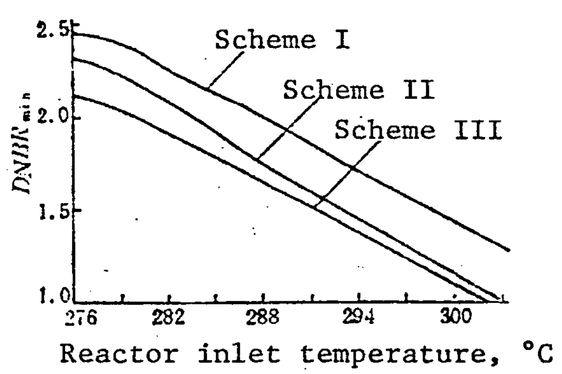

Figure 14. Minimum Burnout Ratio With Changes in Reactor Inlet Temperature

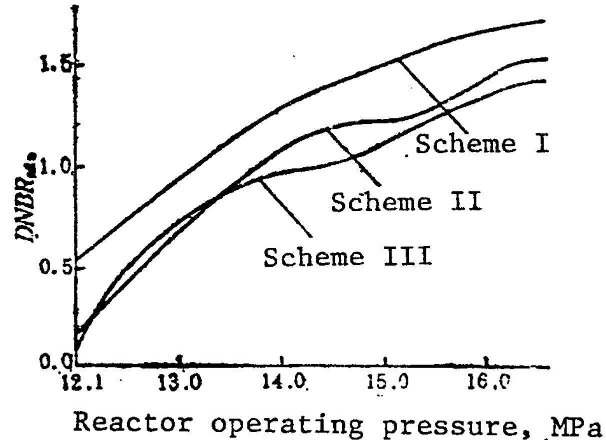

Figure 15. $DNBR_{min}$ at Different Reactor Operating Pressures

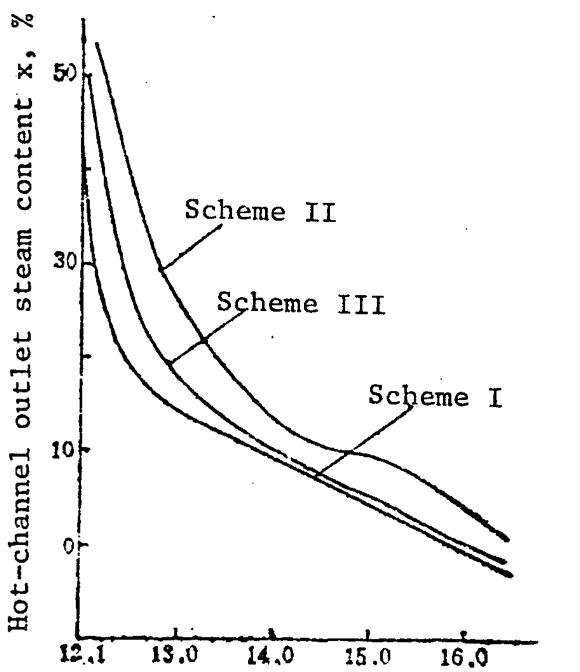

Figure 16. Hot-Channel Outlet Steam Content at Different Reactor Operating Pressures

IV. Shielding Design

1. Design Criteria

(1) To ensure that the reactor pressure vessel does not receive excessive radiation during its entire lifespan, under conditions of a load factor of 0.75 during the specified pressure-vessel lifespan of 60 years, the fast neutron integral flux on its interior surface is less than $1.5 \times 10^{19} n/cm^2$ and the fast neutron integral flux on the interior surface of the suspension basket is less than $1 \times 10^{21} n/cm^2$.

(2) Shielding design should be done according to the rated power and long-term operating conditions. The cost of shielding construction should be reduced with a prerequisite of ensuring irradiation safety. The safety coefficient for the shielding design is assumed to be 2.

2. Results of Computations and Analysis

The shielding-design calculations were completed using a one-dimensional discrete coordinate program ANISN. The quadrature number in the calculations was assumed to be $4(S_4)$ with P_3 expansion. The nuclear data was selected from a 46-group sectional database (25 groups of neutrons and 21 groups of gamma rays).

To reduce the fast-neutron integral flux on the interior surface of the pressure vessel, stainless-steel rod assemblies were used as a core reflecting layer. A metallic reflective adiabatic layer was used with stainless steel 0.03 mm thick in a total of 25 layers and with an outer-wall thickness of 1.5 mm installed equidistantly in 7.5 cm air gaps. The basket material was 0CR18Ni10Ti and was 4 cm thick. The pressure-vessel material was A508-3 steel, 22 cm thick.

Stainless-steel rod assemblies were used as the reflecting layer. Because of the weakened effects of inelastic scattering of the iron on more than 15 groups of fast neutrons, there was an obvious decline in the fast-neutron integral flux of the basket and pressure vessel. This can reduce the fast-neutron integral flux on the interior surface of the pressure vessel by 50 to 60 percent compared to a water-reflecting layer. See Table 4 for more detailed information.

Table 4. Fast-Neutron (E ≥ 1 MeV) Flux on Inner Surfaces of Suspension Basket and Pressure Vessel at Different Iron-Water Volume Ratios

Reflecting-layer iron-water volume ratio	Inner surface of suspension basket		Inner surface of pressure vessel	
	Flux, n/cm^2/s	Integral flux, n/cm^2	Flux, n/cm^2/s	Integral flux, n/cm^2
Pure-water reflecting layer	$7.035x10^{11}$	$9.984x10^{20}$	$9.676x10^{10}$	$1.373x10^{20}$
70%Fe/30%H_2O	$5.703x10^{11}$	$8.093x10^{20}$	$3.678x10^{10}$	$5.219x10^{19}$
80%Fe/20%H_2O	$6.828x10^{11}$	$9.689x10^{20}$	$3.667x10^{10}$	$5.194x10^{19}$
90%Fe/10%H_2O	$9.099x10^{11}$	$1.2912x10^{21}$	$3.954x10^{10}$	$5.611x10^{19}$

The iron-water volume ratio in the stainless-steel reflecting layer has definite effects on the fast-neutron integral flux on the interior surface of the pressure vessel. With an iron volume component between 0.6 and 0.8, the fast-neutron integral flux on the interior surface of the pressure vessel is an extremely small value.

To satisfy presently stipulated design criteria, the thickness of the water layer between the basket and pressure vessel should be increased by about 10 cm, meaning that the internal diameter of the pressure vessel must be increased.

Passive Safety System Design

906B0020C Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 10 No 5, Oct 89 pp 19-23 [MS received 22 May 89]

[Article by Bai Ping [2672 1627] and Tan Zuo [6223 4373] of the Southwest Center for Reactor Engineering Research and Design, Chengdu: "AC-600 Passive Safety System Design"]

[Text] **Abstract**

This article describes the design program and design characteristics of AC-600 passive safety systems, the operation of these safety systems under various accident conditions, and a comparison of its safety and reliability with special safety facilities in existing PWR nuclear power plants.

Key words: passive safety system, residual heat removal, safety injection, loss-of-water accident.

I. Introduction

The design goals for the AC-600 are substantial improvements in safety, reliability, repairability, investments and operating costs, construction schedules, and other areas compared to existing PWR nuclear power plants. One important measure adopted to meet these goals is the use of passive safety systems.

The safety functions implemented in AC-600 passive safety systems include the following transients and accidents:

1. Emergency core residual heat removal (RHR).

2. Compensation for abnormal leakages of reactor coolant, water injection in reactor-coolant-pressure marginal LOCA's, and coolant recirculation.

3. To remove core residual heat into the environment for containment-vessel cooling after pressure drops and accidents, and to prevent radioactive materials from leaking into the environment.

II. Emergency Core RHR System

The main function of the emergency core RHR system is to guarantee removal of the residual heat from the core and maintain the reactor temperature within specified safety limits in the event of damage to the core cooling chain and dissipate the heat into the environment when the reactor loses normal cooling measures in the reactor-coolant system (RCS)—steam generators—secondary-loop system—steam-turbine power [system] and condenser.

These types of accidents constitute typical working conditions under which the emergency core RHR system begins operation:

1. Loss of power to the entire plant.

2. Loss of primary and auxiliary feedwater in the two loops.

3. Rupturing of the main steam pipes.

A simplified flow chart of the emergency core RHR system is shown in Figure 1. There is an independent emergency core RHR system composed of an emergency feedwater tank, an emergency air cooler, and the associated pipes, valves, and control instruments. Each steam generator is fitted with an identical system. The main design parameters of the emergency core RHR system are listed in Table 1.

Table 1. Main Design Parameters for Emergency Core RHR System

Parameter	Value
Emergency feedwater tank	
Operating pressure, MPa	5.88
Design capacity, m^3	25
Height of configuration	4 m higher than steam-generator feedwater pipe
Emergency air cooler	

Table 1. Main Design Parameters for Emergency Core RHR System (Continued)

Parameter	Value
Operating pressure, MPa	5.88
Operating temperature, °C	275
Heat transfer area, m^2	750
Height of configuration	4 m higher than steam-generator feedwater pipe

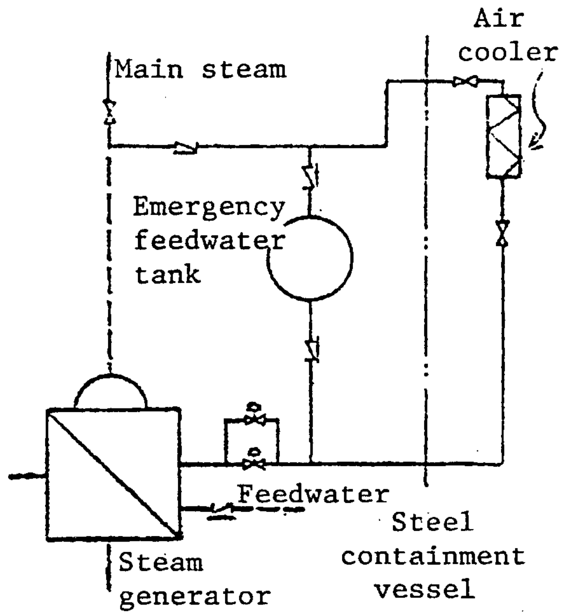

Figure 1. Emergency Core RHR System Flow Chart

When there is a loss of power to the entire plant (obviously, this would interrupt the secondary-side feedwater in the steam generator), the emergency-feedwater-tank isolation valve is opened by a steam-generator low-water-level or feedwater-interrupt signal and supplies the steam generator with a make-up water source to hold the water level on the secondary side of the steam generator above a specified level. The secondary-side water of the steam generator absorbs heat from the reactor coolant and is converted to steam, which flows upward to the emergency air cooler, which condenses the steam into water and transfers the heat to the atmospheric environment. The condensed water flows by gravity back into the steam generator, thereby establishing a continuous natural circulation flow. The emergency core cooling system also establishes corresponding natural circulation because of cooling by the secondary side of the steam generator, thereby attaining the goal of continuous removal of the core heat into the environment.

Similarly, when there is a rupture in the main steam pipe, the emergency feedwater tank will compensate for the loss of secondary-side water in the steam generator caused by steam spraying before the main steam isolation valve is closed, maintaining the water level in the steam generator and establishing natural circulation cooling in the entire system similar to that when a full plant-power-cutoff accident occurs.

The most obvious characteristic of the AC-600 emergency core RHR system is that the steam generator make-up water and cooling loops are composed entirely of passive equipment. Its operation relies totally on steam, convection, gravity, and other natural laws. It does not require an emergency power source, cooling water, or other auxiliary equipment, which greatly improves the operational reliability of the system, and means that operating times are not restricted by water sources, time, or other conditions, so it can carry out long-term core residual heat removal.

III. Safety Injection System

The AC-600 safety injection system mainly provides compensation for abnormal leakage of coolant and water injection and recirculation into the core when a reactor coolant pressure marginal LOCA occurs.

Figure 2 shows a simplified flow chart of the AC-600 safety injection system. It is similar to existing PWR power-plant safety injection systems. This system is also divided into high-, moderate-, and low-pressure injection systems and the associated recirculation systems. The high-pressure injection system portion here is composed mainly of two core make-up water tanks at the same pressure as the reactor coolant. The moderate-pressure injection portion is composed of two safety injection tanks with an operating pressure of about 5.2 MPa. The low-pressure injection and recycling portions are mainly composed of two water pumps located in the

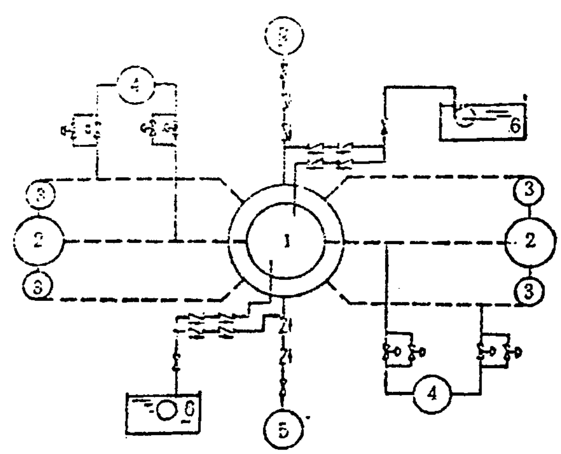

Figure 2. Safety Injection System Flow Chart

Key: 1. Reactor 2. Steam generators 3. Main pumps 4. Reactor make-up water tanks 5. Safety injection tanks 6. Containment-vessel sumps

containment vessel and a containment vessel sump, as well as the associated pipes, valves, and control instruments.

Table 2 lists the main design parameters for the AC-600 safety injection system.

Table 2. Primary Design Parameters for Safety Injection System

Parameter	Value
Core make-up water tank	
Operating pressure, MPa	15.8
Design volume, m^3	40
Medium	Deionized water containing 1300-2100 ppm boron
Safety injection tanks	
Operating pressure, MPa	5.2
Total volume, m^3	60
Filled water volume, m^3	40
Medium	Deionized water containing 1300-2100 ppm boron
Safety injection/recirculation pumps	
Type	Vertical submersible pump
Design flow rate, kg/s	142
Design lift, MPa	1.05
Closed lift, MPa	1.25
Maximum operating temperature, °C	125
Containment-vessel sump	
Normal filled water volume, m^3	550
Maximum filled water volume, m^3	730

When a LOCA due to abnormal leakage or a small rupture occurs in the RCS, the core make-up water tank located at a higher elevation receives a low-water-level signal from the pressure stabilizer and opens the outlet isolation valve. The water in the tank, which contains boron, relies on gravity for injection into the RCS. There is a rather small pipe at the top of the core make-up water tank which carries steam to the steam space of the pressure stabilizer to maintain the internal pressure in the core make-up water tank.

When an accident involving a large rupture occurs in the RCS, the core make-up water tank cannot inject enough water into the reactor to maintain the feedwater amount in the RCS and the system pressure continues to drop. When it falls to about 5.2 MPa, the boronated water in the safety injection tank is pushed into the RCS by nitrogen gas in the upper part and floods the core. In addition, a pipeline with a rather large-diameter pipe connects the cold end of the main pipe channel with the top of the core make-up water tank. In this way, a pressure drop in the RCS caused by a LOCA from a large rupture removes the steam generated in the RCS through the cold end to the top of the core make-up water tank; this causes the fluid within the tank to be pushed by the steam in a rather large flow which is injected into the core.

Because the water volume of the safety injection tank and core make-up water tank is very limited, when there is a loss of coolant from a break at both ends of the main pipe, the safety injection tank is emptied within 100 seconds and the core make-up water tank is emptied within 300 seconds. The RCS pressure continues to drop. When it falls below the sealed pressure head of the safety injection pump, the safety injection pump injects the water absorbed into the containment-vessel sump into the RCS and further floods and cools the core.

The reactor coolant flowing out of the rupture in the RCS collects in the containment-vessel sump and is returned by the safety injection pump into the RCS, thereby forming long-term core circulation cooling.

The AC-600 safety injection system has two obvious advantages compared to special safety equipment in existing PWR power plants:

1. Active high-pressure safety injection is replaced by a passive full-pressure core make-up water tank. This saves the high-pressure safety injection pumps which are hard to manufacture and cost a great deal, as well as the mechanical instrument systems which serve these pumps. Moreover, in this part of the system, excluding the need for a DC power source to operate the signals and valves, the remainder can rely on natural gravity to perform the expected functions, which greatly improves system reliability. In addition, it avoids certain situations in which the RCS pressure is higher than the closed pressure head of the safety injection pump; this prohibits operation of the safety injection pump in the event of a water loss accident due to a small rupture, which increases system operability.

2. The water sources for both long-term low-pressure safety injection and core injection come from the containment-vessel sump, which avoids the conversion of the water source from the exchange water tank to the containment vessel sump from the injection to the recirculation stage in existing PWR power plants. This reduces the likelihood of operating mistakes. Moreover, because all the safety systems in the AC-600 are located in the containment vessel, whenever failure or leakage occurs and the system begins operation, radioactive materials do not leak out from the containment vessel to pollute the environment.

IV. Containment-Vessel Cooling System

As mentioned previously, whenever an accident occurs at the power plant, relying on safety injection in the AC-600 can enable long-term core-coolant recirculation to remove residual heat from the core. However, by relying only on the safety injection system for operation, the residual heat in the core always accumulates within the containment vessel. The air temperature in the

vessel, including the water temperature in the containment-vessel sump, becomes very high, which makes it impossible to cool the core. Moreover, because much steam would leak into the containment vessel in extremely limited accident conditions in which there is a rupture at both ends of the main pipe and the main steam pipe ruptures, this would cause a rapid increase in the air temperature and pressure within the vessel. To prevent damage to the integrity of the containment vessel, the air within the containment vessel also must be cooled. For this reason, the AC-600 has a passive containment-vessel cooling system. This system has two main functions:

1. To prevent the air temperature and pressure within the containment vessel from exceeding the design limits, assure the integrity of the containment vessel, and prevent leakage of radioactive materials into the environment.

2. To conduct the residual heat in the core into the environment in the event of a loss of normal cooling in the reactor due to a LOCA and prevent a core meltdown.

The AC-600 containment vessel cooling system is illustrated in Figure 3.

When a large rupture in the RCS causes a LOCA, a large amount of steam can escape from the rupture and spray into the containment vessel, raising the temperature and pressure in the vessel. The isolation valve for the water storage tank located on top of the containment vessel opens automatically after it receives a high-pressure

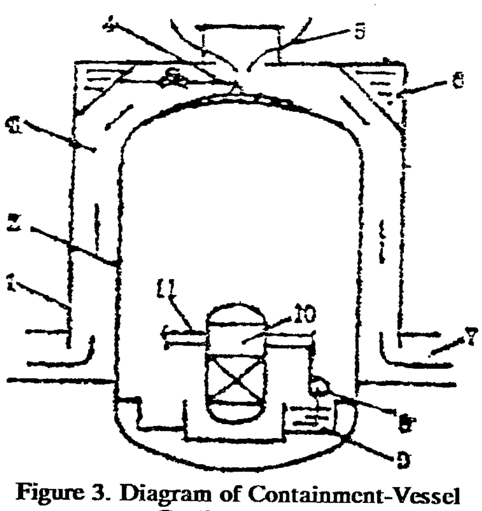

Figure 3. Diagram of Containment-Vessel Cooling System

Key: 1. Concrete containment vessel 2. Steel containment vessel 3. Containment-vessel interlayer 4. Containment-vessel coolant-water sprayer 5. Containment-vessel interlayer ventilation outlet 6. Containment-vessel cooling-water storage tank 7. Containment-vessel interlayer ventilation inlet 8. Safety injection pump 9. Containment-vessel sump 10. Pressure vessel and reactor core 11. Main pipeline

signal from the containment vessel. The cold water sprays onto the outer surface of the containment vessel, which has an internal steel surface, and cools the steel containment vessel. At the same time, the energy in the air in the containment vessel is transferred to the environment through the steel containment vessel, which reduces the air temperature and pressure within the vessel. About 3 days after the water begins spraying, the containment-vessel cooling-water storage tank will empty, after which it relies on a natural ventilation convection formed by the interlayer between the concrete containment vessel and the steel containment vessel for long-term cooling of the air within the containment vessel.

At the same time, when a LOCA occurs in the RCS, the residual heat in the core is carried into the air in the containment vessel by the liquid flowing out of the rupture. Thus, cooling the air in the containment vessel also removes the residual heat from the core and cools the core.

The main characteristics of this system are that it uses a containment vessel with an internal steel layer as a heat-exchange surface to cool the air within the containment vessel and remove residual heat from the core by employing entirely passive equipment. This eliminates the containment vessel spray system in existing PWR power plants and does not require any other intermediate circulating cooling medium like equipment cooling water, sea (river) water, and so on. Moreover, it uses the passive equipment directly to remove the residual heat in the core into the environment, which conserves investments and increases system operation reliability.

V. Conclusion

Substantial improvements were made in the AC-600 safety system compared to the special safety systems in existing PWR power plants. On the one hand, passive equipment is employed, which increases system operation reliability. Preliminary analysis[1] indicates these systems can satisfy the requirements of guaranteeing power-plant safety under all types of accident conditions. On the other hand, because it eliminates most active equipment in existing PWR nuclear power plants which have strict requirements and are difficult to manufacture, such as auxiliary feedwater pumps, high-pressure safety injection pumps, containment-vessel spray pumps, and so on, the AC-600 safety system offers the possibility of domestic production. At the same time, eliminating this equipment along with support measures serving it greatly simplifies the system.

References

1. Zhang Senru [1728 2773 1172] and Tan Zuo [6223 4373], "Behavior Analysis for the Passive Safety Systems of AC-600," in IAEA TECHNICAL COMMITTEE MEETING, Moscow, March 1989.

Analysis of Passive Safety Features

906B0020D Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 10 No 5, Oct 89 pp 24-29, 48 [MS received 11 Mar 89]

[Article by Zhang Senru [1728 2773 1172], Tan Zuo [6223 4373], Zhang Zongyao [4545 1350 5069], Wang Jianyu [3769 1696 3254], Bai Ping [2672 1627], and Li Dongsheng [2621 0392 3932] of the Southwest Center for Reactor Engineering Research and Design, Chengdu]

[Text] **Abstract**

The research focus in the conceptual design for the AC-600 includes the core and passive safety systems as well as system simplification. This article discusses the natural-circulation capabilities and performance of the passive safety systems during periods of LOCA or full plant power outages. It also provides a preliminary discussion of passive restrictions on excess reactivity and thermal hydrodynamic transients.

Key words: AC-600, passive safety systems, safety evaluation.

I. AC-600 Passive Safety Goals and Measures

Two primary safety goals were proposed in the conceptual design of the AC-600: 1) A core-meltdown frequency of 1 x 10^{-5} to 1.5 x 10^{-6}/reactor-year. 2) A radiation dosage of 0.5 to 1 Sv/year for nuclear power plant personnel. Five primary measures were adopted to meet these safety requirements:

1. Reducing excessive reactivity in the core, lowering the probability of transient criticality accidents occurring, and reducing serious consequences of reactivity accidents.

2. Employing full-pressure core make-up water tanks and sufficiently large safety injection tanks to assure that boronated water floods the core quickly after a LOCA.

3. Increasing the vertical distance between the steam generator and core, reducing the flow resistance in the main loop, and improving natural-circulation capabilities.

4. Using emergency feedwater tanks and air-cooled heat exchangers, and relying completely on natural-circulation cooling in the first and second loops to remove residual heat from the core.

5. Passive natural-circulation cooling to remove residual heat from the core.

Nuclear power plant safety includes environmental safety and the safety of the nuclear power plant itself. These two aspects are closely related. The accident at Three Mile Island in the United States did not cause radioactive pollution of the environment or endanger public health, which satisfied the requirement for environmental safety of the nuclear power plant. There was, however, serious damage to the nuclear power plant, which resulted in enormous economic losses to the proprietors. The public's trust of nuclear power plants did not rise because of the first aspect. Instead, their doubts and concern rose because of the latter aspect. For this reason, the AC-600 design treated environmental safety and safety of the reactor itself as the two most important safety goals. Table 1 lists the main design parameters for the AC-600.

Table 1. AC-600 Primary Design Parameters

Reactor power, MW_t	1820
Number of loops	2
RCS operating pressure, MPa	15.5
Reactor inlet temperature, °C	287.7
Reactor outlet temperature, °C	324.5
Reactor-coolant total flow rate, kg/s	4444.4
RCS total resistance, MPa	0.517
Steam-generator secondary-side pressure, MPa	5.88
Steam output per steam generator, kg/s	501
Linear power density of core elements, kW/m	12.3
$DNBR_{min}$	2.002

II. Passive Safety in a Transient Increase in Reactor Power

To reduce the probability of a core meltdown due to a reactivity accident in existing water-cooled-reactor nuclear power plants, the core should have a small amount of excess reactivity and sufficiently large negative-reactivity temperature coefficients. When an accident occurs in a nuclear power plant, if the core has excess reactivity, it is entirely possible for the temperature rise to induce counteraction by negative reactivity prior to a nuclear fuel meltdown. This gives the reactor a greater capacity for avoiding core meltdown accidents caused by transient criticality.

The AC-600 employs Gd_2O_3 burnable poison and compressed-water-rodspectral-shift operation. In the initial period of the fuel cycle (prior to 70 percent), the compressed-water rods are inserted into the core. The water-uranium ratio is 2.042, the energy spectrum hardens, k_{eff} is reduced, and more ^{238}U is converted into fissionable nuclei. In the later period of the fuel cycle (after 70 percent), the compressed-water rods are pulled out of the core, the water-uranium ratio changes to 2.369, the energy spectrum softens and k_{eff} increases. This can extend the fuel cycle time by about 10 percent, as shown in Figure 1.

During the operating process, excess reactivity in the AC-600 core is much smaller than in existing 600 MW_e PWR nuclear power plants. A comparison can be made in Figure 2. The critical boron concentration during the operating process is about 100 ppm, equivalent to 1 percent reactivity. A smaller critical boron concentration can provide a larger negative-reactivity temperature coefficient. After the group of control rods with the

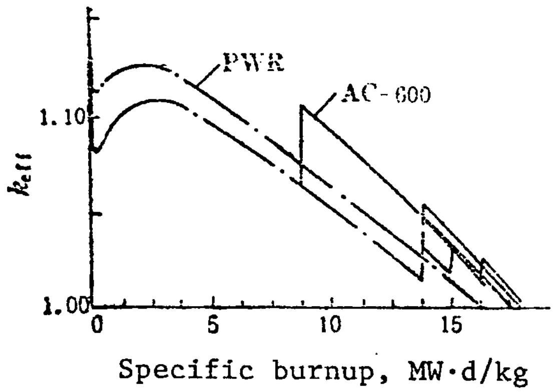

Figure 1. Effects of Compressed-Water-Rod Operation Patterns on Effective Neutron Multiplication Coefficient k_{eff}

highest value is removed, the control-rod reactor-shutdown margin at hot-state zero power is 2.5 percent (see Figure 3 in Zhang Zongyao [4545 1350 5069] et al., "AC-600 Core Design" in this issue of HE DONGLI GONGCHENG).

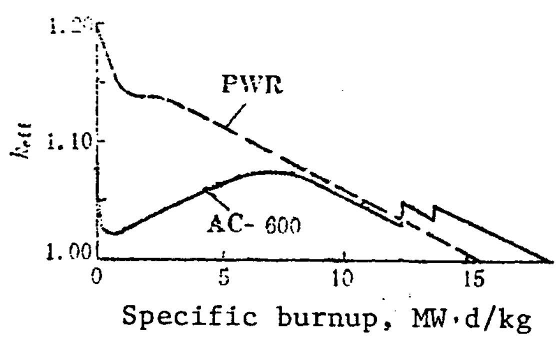

Figure 2. k_{eff} After Adoption of Gd_2O_3 Burnable Poison

It is apparent that improved passive safety and inherent safety during abrupt increases in reactor power caused by inherent reactivity accidents in a reactor give the core a smaller excess reactivity and a larger negative-reactivity temperature coefficient. This is one of the characteristics of the AC-600 design.

III. AC-600 Natural-Circulation Cooling Capabilities

The relative vertical height from the core outlet to the pressure-vessel outlet in the AC-600 is 4.99 m. The relative vertical height from the pressure-vessel outlet to the steam-generator (SG) heat-transfer pipe inlet is 5.58 m. Table 2 graphs the pressure drop at each end of the main coolant loop when there is a total cessation of operation in the reactor-coolant circulation pumps for a reactor thermal-power rated value of 25 percent. The natural-circulation coolant flow rate is 4,852 tons/hour, which is 15.12 percent of the rated value. Table 3 provides thermal data for the core under natural-circulation coolant operating conditions at different power levels in the AC-600 reactor. The data show that: 1) When the reactor power increases, there is also a corresponding increase in the natural-circulation flow rate. 2) DNBR is rather large. 3) The heat-channel outlet steam quality (X^h_o) increases as the core power increases. In consideration of the stability and safety of the core current, the AC-600 can rely on natural-circulation cooling to remove the power at 25 percent of the rated power, which greatly improves the passive safety performance of the AC-600.

Table 2. Pressure Drops in Each Stage of the Natural-Circulation Primary Coolant System at 25 Percent of Rated Core Power (MPa)*

Core	.0273
From core outlet to pressure-vessel outlet	.0321
From pressure-vessel outlet to SG heat-transfer-tube inlet	.0382
SG heat-transfer-tube segment	.0097
From SG heat-transfer-tube outlet to main pump	-.0412
From main pump to pressure-vessel inlet	.0003
From pressure-vessel inlet to core inlet	-.0664

*In the table, positive values are pressure drops, negative values are pressure rises.

Table 3. Core Parameters for Natural-Circulation Cooling Conditions

Rated core power	0.10	0.15	0.20	0.25	0.30	0.35
Reactor outlet coolant temperature, °C	307.7	314.1	318.9	324.4	328.7	332.9
Reactor coolant flow rate, t/h	3561	4065	4517	4852	5186	5446
$DNBR_{min}$	16.94	11.22	8.41	6.57	5.09	4.19
Hot-channel outlet steam quality, X^h_o	-0.156	-0.094	-0.042	0.016	0.122	0.173

IV. Main-Pump Coasting Requirements and Power-Outage Accident Analysis

The AC-600 uses a sealed main pump. Because the fully sealed pump does not have a coasting [i.e., idling] flywheel, the rotational inertia is very small. The main-pump design must stipulate minimum demands for the main pump coasting flow rate to prevent endangering core safety when there is a main-pump power outage.

Computations and evaluations can be made using the time range of half-flow rates listed in Table 4, and analysis of Figures 3 to 5 shows that the faster the reduction in main-pump flow rate coasting, the smaller the DNBR and the higher the X^h_o. The curves numbered 1 to 6 in Figures 3 to 5 correspond, respectively, to the half-flow rate times of 2.5 to 12.5 seconds in Table 4.

Table 4. Core Hot-Channel DNBR and X^h_o During Power Cutoff to Main Pumps

Half flow-rate time, s	2.5	4.0	6.0	7.0	10.0	12.5
Transient minimum DNBR	1.57	1.77	1.82	1.84	1.87	1.89
Transient maximum X^h_o	0.286	0.104	0.067	0.056	0.038	0.029

In consideration of fluctuations in the main parameters, instrument measurement error, and inexactness in the calculations, the AC-600 main-pump coasting half-flow rate time should be greater than 4.0 seconds. An electrically powered generator with a large flywheel connected to the main pump can be used. During normal operation, the electrically powered generator operates at no load, but continues coasting when there is a power shutoff to provide electricity to the main pump, extend the main-pump coasting time, and satisfy safety requirements.

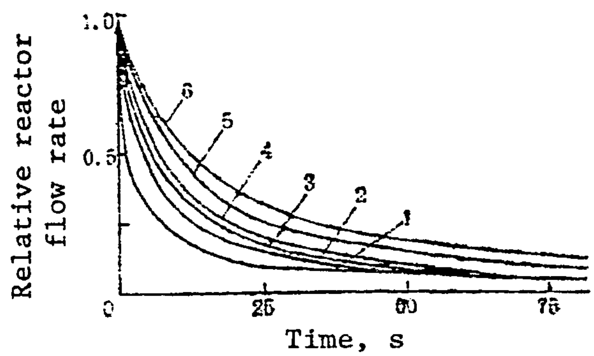

Figure 3. Coasting Flow Rate After Power Cutoff to Main Pumps

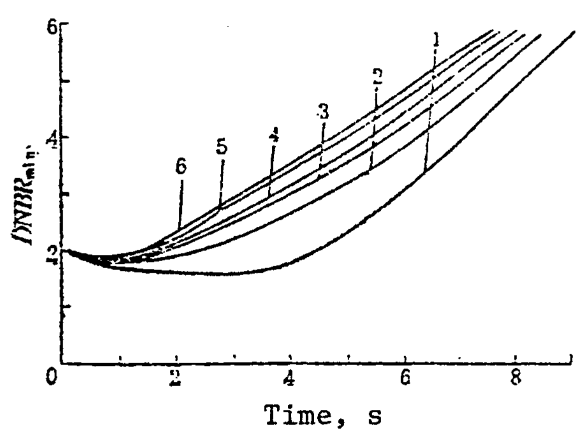

Figure 4. DNBR After Power Cutoff to Main Pumps

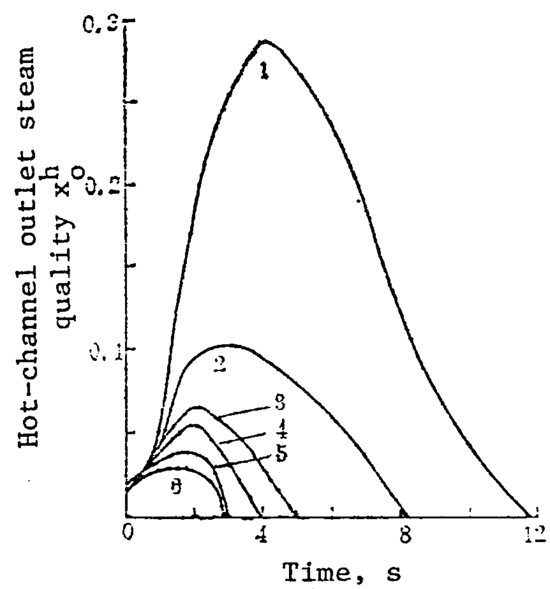

Figure 5. Core Hot-Channel X^h_o After Power Cutoff to Main Pumps

V. Emergency Residual Heat-Removal System (ERHRS)

This system mainly replaces the normal second-loop system under accident working conditions like a rupture in the main steam pipes, loss of primary feedwater, and so on when there is a full plant power outage. It removes the residual heat in the core and maintains the coolant temperature and pressure in the first loop and the pressure in the secondary side within permissible limits.

The system flow chart is shown in Figure 6. After a loss of main feedwater or rupture in the main steam pipes, the water level and pressure in the secondary-side steam generator drop. The water in the emergency feedwater

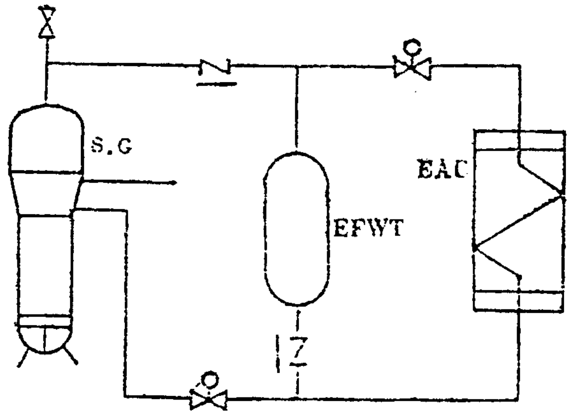

Figure 6. Diagram of Emergency Residual Heat-Removal System

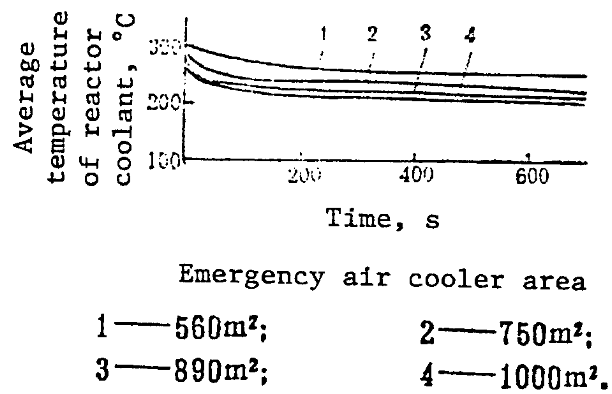

Figure 7. Average Temperature of Reactor Coolant After Full Plant Power Shutoff

tank (EFWT) flows under the effects of gravity down to the secondary side and absorbs the heat transferred from the primary side. As it is converted to steam and rises, it flows into the emergency air cooler (EAC) and the steam is condensed there, after which it flows into the steam generator, forming a natural circulation which provides a continual supply of air to the core decay heat zone.

The most important aspect of this design is that it relies entirely on natural circulation to establish a cooling loop. With the exception of a few valves which depend on storage batteries for their drives, no other power sources for motive power are required.

The design volumes of the two main pieces of equipment in the system, the EFWT and EAC, are related to the thermal hydrodynamic behavior of the first and second loops in the power unit.

The design basis for determining the volume of the EFWT was to provide an adequate water level on the secondary side of the steam generator when there is a rupture in the main steam pipe. Analysis indicated that this capacity should be about 25 m^3.

The design working conditions for the EAC were a full plant power outage and loss of primary feedwater. Different air cooling heat-transfer areas were selected here for analysis of the thermal hydrodynamic behavior of the first and second loops. The computed results are shown in Figure 7.

In the conceptual design of the AC-600, it was assumed that the heat-transfer area of the EAC was 750 m^2. The respective natural-circulation flow rates for the RCS and secondary side were calculated on this basis. The results are shown in Figure 8. The computations show that the primary and secondary-side natural-circulation flow rates could be maintained at about 5 percent and 3.5 percent of the full-power-operation flow rates for a rather long period of time.

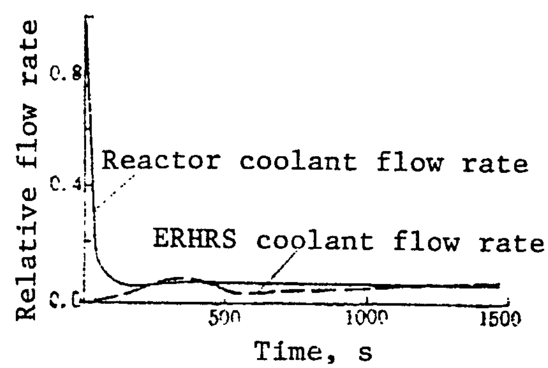

Figure 8. Natural-Circulation Flow Rate of Reactor Coolant and ERHRS Coolant After Full Plant Power Shutoff

Analysis of the results shows:

1. From the perspective of thermodynamics, selection of appropriate system design parameters is sufficient for using the ERHRS to cool the core by natural circulation.

2. The ERHRS system volume required to guarantee that the second-loop system (the secondary side of the steam generator) does not exceed pressure requirements is greater than the volume required to guarantee that the first-loop-system coolant does not exceed the operating values. Thus, not exceeding the pressure in the secondary side is the basis for the ERHRS design.

3. Increasing the design volume of the ERHRS can shorten the time required to reduce the reactor cooling temperature. These systems, however, are safety devices which are used extremely little. From economic considerations, too large a capacity should not be chosen. Too large a volume would also cause the rate of temperature drop in the first and second loops during the early stages of system operation to be too large, which would create a rather large temperature shock to the equipment.

VI. Safety Injection System

The AC-600 safety injection system (SIS) is illustrated in Figure 9. It is composed of two core make-up water tanks (CMT), two safety injection tanks, and two low-pressure safety injection/recirculation pumps located in the containment-vessel sump. Table 5 and other design data were used to analyze a LOCA involving a break in both ends of the cold leg of the main pipe. The main results are shown in Figures 10 to 13. The results show:

1. The flow rate from the CMT can be sustained for a rather long time, and it can quickly flood the core.

2. Under large LOCA conditions, the CMT flow rate is greater than the high-pressure flow rate in the conventional design. When there is a small rupture, the injection flow rate is somewhat smaller. The CMT has a passive design and is much more reliable than a high-pressure injection pump.

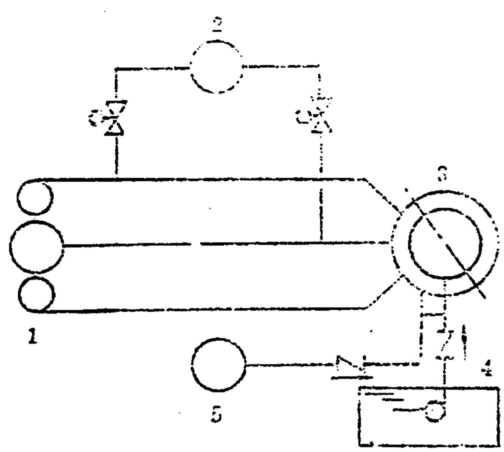

Figure 9. Diagram of AC-600 Safety Injection System

Key: 1. Main pump 2. CMT 3. Reactor 4. Containment-vessel sump 5. Safety injection tank

Table 5. Data on Primary Equipment in Safety Injection System

CMT operating pressure, MPa	15.8
CMT volume, m^3	40x2
Safety-injection-tank operating pressure, MPa	5.2
Safety-injection-tank water volume, m^3	40x2
Low-pressure safety-injection/recirculation pump	
Design flow rate, kg/s	142
Low-pressure safety-injection/recirculation pump	
Design pressure head, MPa	105
Containment-vessel sump water volume, m^3	550x2

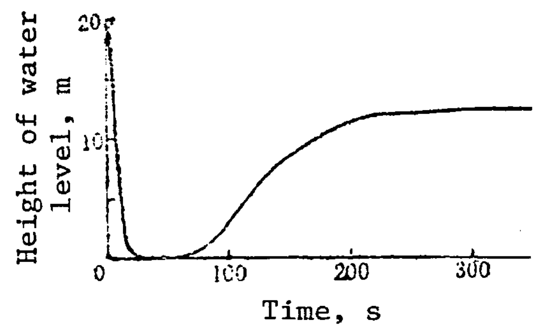

Figure 10. Water Level in Reactor Pressure Vessel After Large-Break LOCA

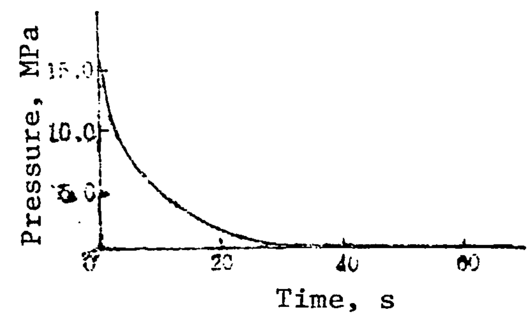

Figure 11. RCS Pressure After Large-Break LOCA

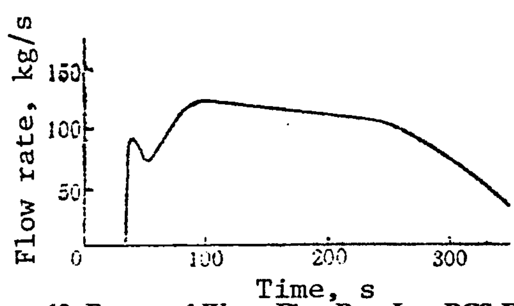

Figure 12. Boronated-Water Flow Rate Into RCS From CMT After Large-Break LOCA

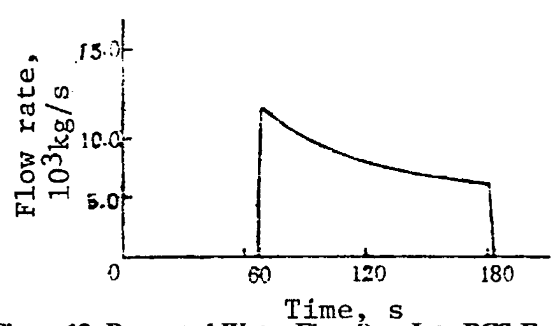

Figure 13. Boronated-Water Flow Rate Into RCS From Safety Injection Tank After Large-Break LOCA

VII. Containment-Vessel Cooling System

The AC-600 has a dual-layer containment vessel with an inner layer that is a steel vessel. In a large-break LOCA or a main steam-pipe rupture accident, when the pressure and temperature within the containment vessel rise, the water kept in the water storage tank at the top of the large vessel, which is the outer layer, is sprayed by the effects of gravity onto the steel vessel to reduce the rising temperature and pressure in the containment vessel. At the same time, the safety injection/recirculation pump located in the sump injects the boronated water in the sump into the RCS. After absorbing the heat in the core, the steam-water mixture which flows out of the rupture is cooled inside the containment vessel and collects in the sump.

Analysis of the results of system cooling effects shows that in the initial stages of an accident, this equipment can effectively reduce the temperature and pressure of the air in the containment vessel. However, as the temperature and pressure in the vessel drop, heat transfer along the surface of the inner and outer walls of the steel vessel degrades, especially after the water spray in the containment-vessel storage water tank ends. When cooling depends entirely on natural circulation of the air in the interlayer, heat transfer along the inner and outer walls of the steel vessel degrades and causes a slowing of the drop in the temperature and pressure in the vessel (see Figures 14 and 15).

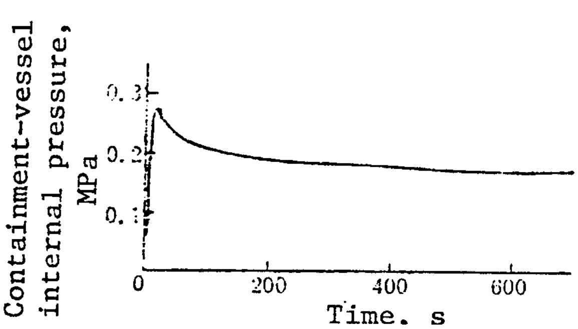

Figure 14. Containment-Vessel Internal Pressure After Large-Break LOCA

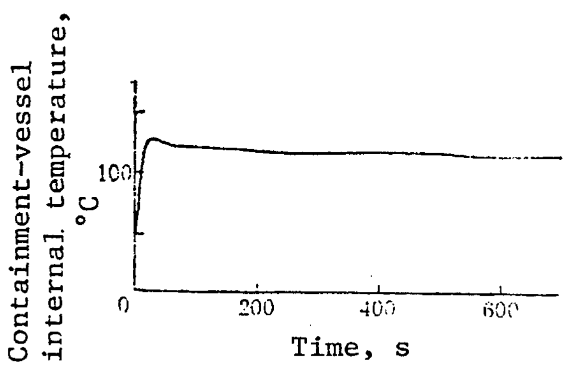

Figure 15. Containment-Vessel Internal Temperature After Large-Break LOCA

For the system, the main thing in the initial stages of an accident is to reduce the pressure within the vessel. In the later stages, it is to remove the residual heat from the core to prevent a rise in the temperature and pressure within the vessel. For this goal, the system can meet requirements.

Conceptual Design of Reactor Structure

906B0020E Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 10 No 5, Oct 89 pp 30-34 [MS received 25 May 89]

[Article by Guan Jianwei [7070 1696 4850] of the Southwest Center for Reactor Engineering Research and Design, Chengdu]

[Text] **Abstract**

This article provides a brief introduction to the conceptual design of the AC-600 reactor, including design conditions for core components, the internal structure of the reactor, the pressure vessel and the drive mechanism.

Key words: inherent safety, spectral-shift control, neutron reflector, calandria structure, single-control-rod disassembly and assembly.

The hot power of the AC-600 reactor is 1,820 MW. Its coolant flow rate is 32,100 tons/hour, the reactor inlet temperature is 287.7°C, the outlet temperature is 324.5°C, the operating pressure is 15.49 MPa, the total height of the reactor is 19.1 meters, the maximum outer diameter is 5.04 meters, and the wet weight of the reactor is about 762 tons. The reactor pressure vessel has two coolant outlets with an internal diameter of 737 mm and four coolant inlets with an internal diameter of 521 mm. These six outlets and inlets link the reactor with the coolant pumps and steam generators to form a sealed primary coolant system. A cross-sectional diagram of the AC-600 reactor is shown in Figure 1.

I. Reactor Pressure Vessel

SA508-3 steel was chosen as the parent material for the pressure vessel. China has already developed SA508-3 steel and matching welding materials. The interior wall of the pressure vessel is butt-welded to a stabilized austenite stainless-steel sleeve. The pressure vessel has an internal diameter of 3.99 meters, is 14.91 meters tall, and weighs 434 tons. The design pressure is 17.16 MPa, the design temperature is 350°C, and the design lifespan is 60 years. The neutron integral flux on the interior wall of the pressure vessel is no greater than $1 \times 10^{19} n/cm^2$ (E greater than or equal to 1 MeV).

There are 137 drive-mechanism tube fittings distributed evenly on the top of the pressure vessel, 52 of them being control-rod magnetic-jack tube fittings and 85 being compressed-water-rod hydraulic-jack tube fittings.

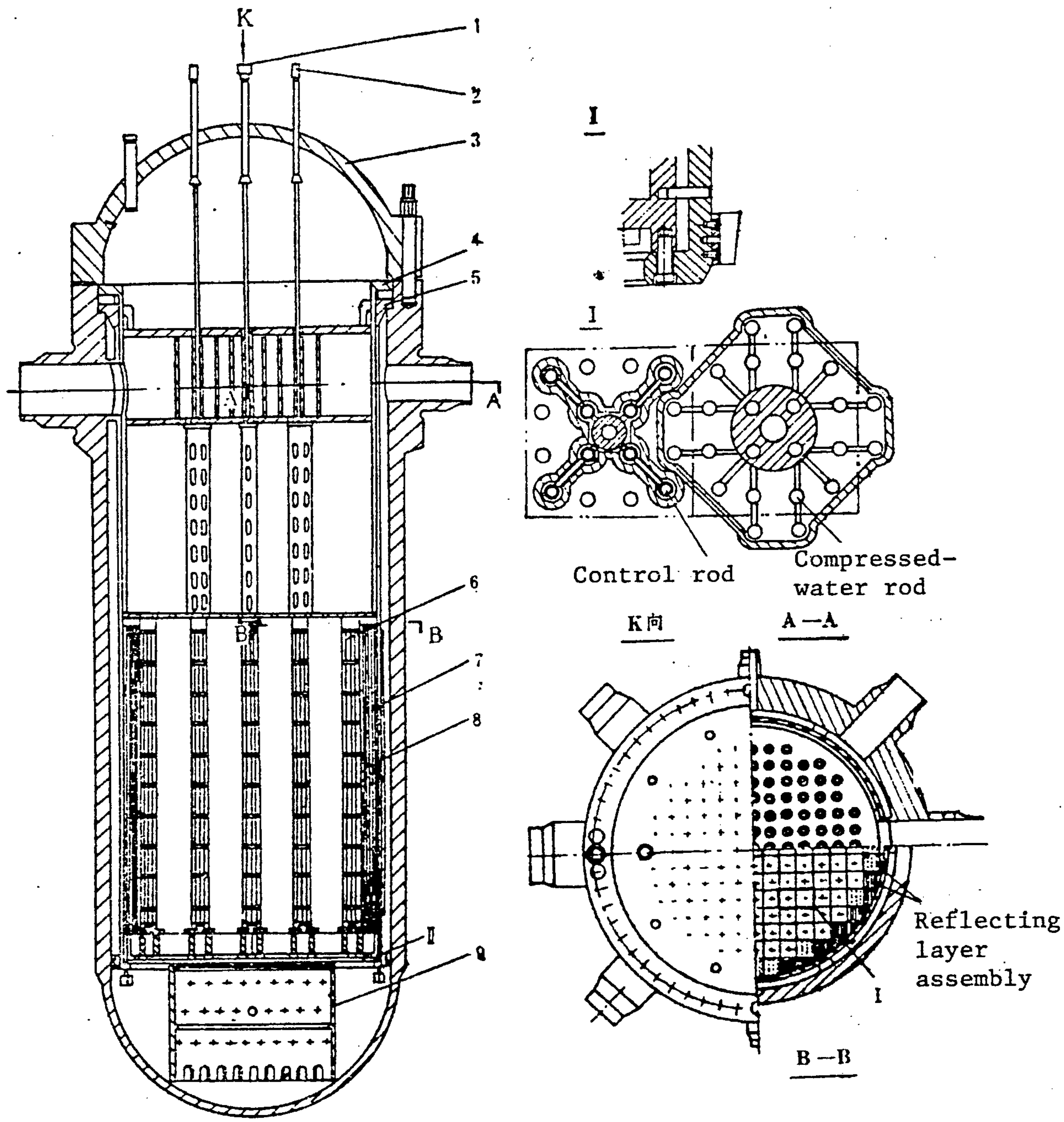

Figure 1. Cross-Sectional Diagram of AC-600 Reactor

Key: 1. Control-rod magnetic-jack tube fitting 2. Compressed-water-rod hydraulic-jack tube fitting 3. Pressure vessel 4. Top assembly 5. Bottom assembly 6. Fuel assembly 7. Reflecting-layer assembly 8. Irradiation monitoring tube 9. Flow-rate distribution skirt

There are eight 100-mm-diameter core-measurement-system exit tube fittings distributed evenly around these tube fittings and two 160-mm-diameter passive safety-injection-system water-injection tube fittings. These permit direct injection of safety injection water into the core outlet to ensure the safety of the reactor core in a LOCA and reduce the probability of an accidental core meltdown. The assemblies on the top are integrated into one unit to facilitate assembly, disassembly, inspection, and maintenance.

Eight radial connecting tubes are distributed evenly on the same planar cross section of the flange below the pressure vessel. Two are reactor coolant outlets and four are coolant inlets. The other two are water-injection ports for the passive safety injection system. The safety injection water can be injected directly into the reactor core through these two tubes, which have openings 200 mm in diameter.

The pressure vessel has these design characteristics: 1) It uses SA508-3 steel, a material in common use internationally. 2) It has an integrally forged flange with a reinforced section with an opening for the connecting tube. This reduces the welding of seams and has good safety characteristics. 3) The barrel is integrally forged and has no longitudinal welded seams or any circular welding seams in the high-neutron-flux region. 4) The pressure vessel inlets and outlets are distant from the core to guarantee the safety of the core during a loss of coolant and reduce the probability of an accidental core meltdown.

II. Core Components

The AC-600 core uses the results of previous scientific research, experiments, designs, and calculations for 17 x 17 assemblies. With a prerequisite of retaining unchanged the rod diameter of the original 17 x 17 assemblies and the dimensions of the grid, the fuel assemblies were designed as 19x19—16(2x2) assemblies. The external dimensions were 240 x 240 x 4,170 mm, and the assemblies weigh 720 kg (see Figure 2).

This assembly uses a bimetallic positioning frame chosen through thermodynamics experiments with upper and lower tube openings that can be disassembled. Sixty-four of the 361 grid positions are occupied by 16 guide thimbles, each guide thimble occupying four grid positions. Another 3 to 5 grid positions are fitted with burnable poison rods and measurement tubes. Fuel rods are installed in the remaining 292 to 294 grid positions. The reactor as a whole is composed of 137 fuel assemblies having an equivalent diameter of 3.17 m and an active section height of 3.66 m.

The equilibrium refueling enrichment of the fuel is 3.2 to 3.3 percent. They are cylindrical UO_2 core blocks 11 mm tall with an external diameter of 8.05 mm. They are assembled into Zr-4 outer-casing tubes with a diameter of 9.5 mm and a wall thickness of 0.64 mm. The charged helium pressure in the tubes is 1.96 MPa. The burnable poison material is Gd_2O_3-UO_2. The guide thimbles have an internal diameter of 22 mm and are made of Zr-4. The strips in the positioning perforated plate are made of Zr-4 and the springs are made of GH-169. The reactor as a whole has 52 control-rod assemblies and each assembly contains eight silver-indium-cadmium rods 20.4 mm in diameter. There are 85 groups of compressed-water rods. The compressed-water rods are 20.4 mm in diameter and [the casing is] made of Zr-4. The configuration of the compressed-water rods and control rods within the reactor is shown in Figure 1.

The structural design characteristics of the reactor are: 1) Cylindrical core blocks are used to reduce the temperature in the center of the fuel. 2) Gd_2O_3-UO_2 is used as the burnable poison to increase the uranium utilization rate and flexibility of the configuration. 3) With the exception of a small amount of GH-169 used for the springs, Zr-4 is used for all the structural material in the core to reduce useless losses of neutrons. 4) The ability to disassemble the upper and lower tube fittings facilitates replacement of damaged fuel rods and improves the economy of the fuel cycle. 5) Compressed water rods are used for spectral-shift control, which lengthens the lifespan of the core by about 60 days. 6) The refueling schedule is 18 months and the average specific burnup of the removed fuel is 55,000 MWxd/t(U).

III. Internal Structure of the Reactor

The internal structure of the AC-600 APWR is composed of an upper assembly, lower assembly, core measurement assembly, secondary support assembly, and so on. There are no heat shields or shroud components. These have been replaced with neutron reflecting assemblies, which increases the core neutron utilization rate and can reduce the neutron integral flux in the pressure vessel. The neutron-reflecting-layer assembly has the same geometric shape as the fuel assemblies and is made of stainless steel. This facilitates installation and adjustment of the internal structure of the reactor.

To meet the guide requirements of the control rods and compressed-water rods, the upper assembly is capable of greater movement than standard internal reactor structures. Two types of specially shaped tube casings were designed as control-rod and compressed-water-rod guides. A group of 24 compressed-water rods is placed across five groups of fuel assemblies. The control-rod assemblies and compressed-water-rod assemblies were designed so that each rod can be assembled and disassembled to facilitate assembly, disassembly and refueling.

To achieve inherent safety in the overall structural design of the AC-600 reactor, the water volume within the core and the height of core flooding were increased. The distance between the core outlet and pressure-vessel outlet is greater than 3.66 m. This provides advantages for the upper assembly within the reactor, and it protects the control rods and compressed-water rods within their guide casings through the entire process of their lifting movements, which prevents lateral shocks from the water current at the reactor outlet. The outlet region of the lower module within the reactor is the supporting structure for the exhaust tubes. The 137 drive rods pass through the tubes in the exhaust-tube support structure and the reactor coolant water flows along the outside of the exhaust tubes out of the reactor into the steam generator.

The internal structure of the reactor has a total height of 10 m, a maximum outer diameter of 4.22 m, and weighs about 293.6 tons. The structure is shown in Figures 1 and 2. The design characteristics are: 1) The heat shield and shroud assemblies are eliminated and replaced with a radial neutron-reflecting-layer assembly. 2) The upper

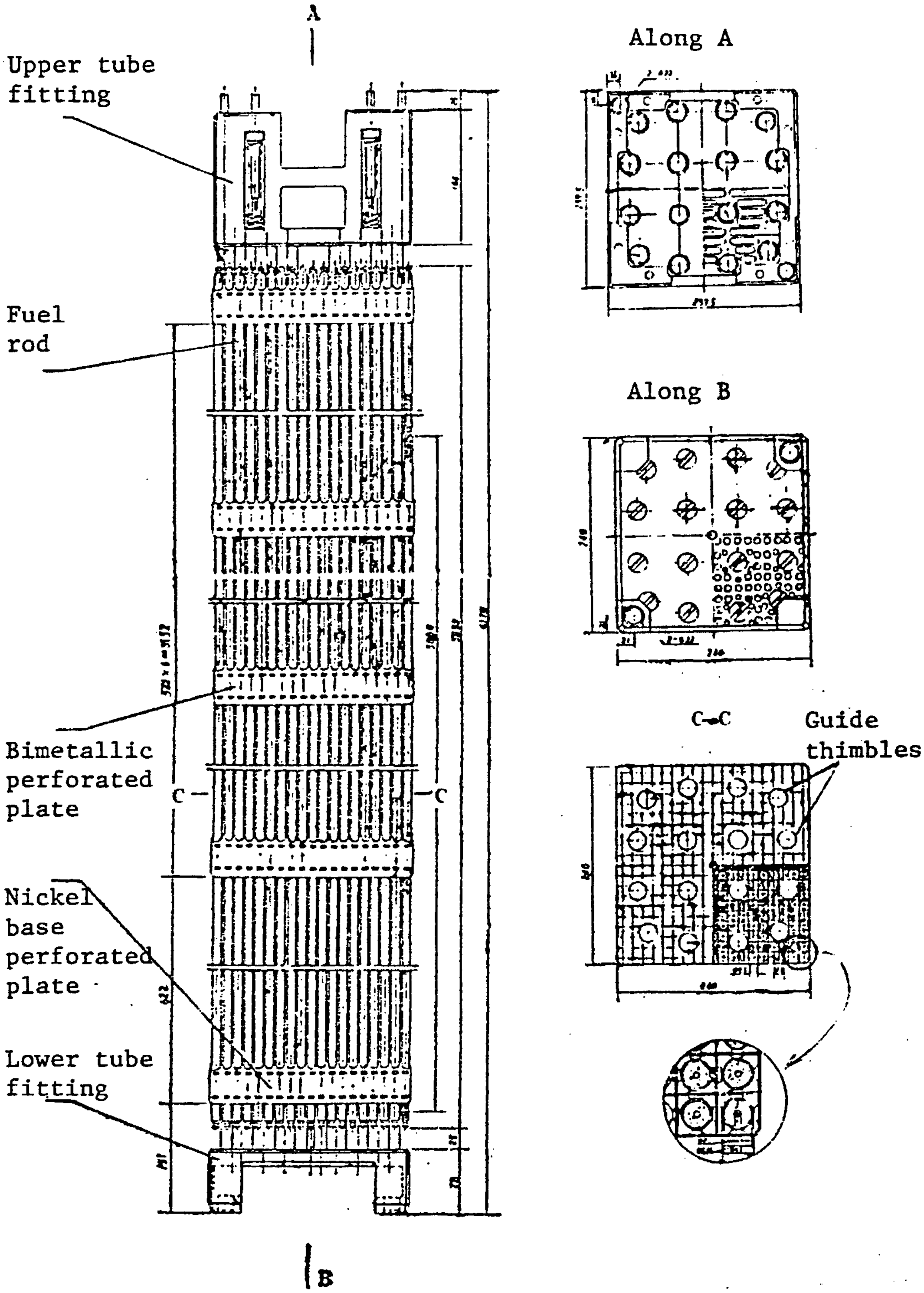

Figure 2. Fuel Assembly

assembly uses the exhaust-tube support structure and specially shaped guide-casing structure to increase the rigidity of the support and help protect the control rods and compressed-water rods. 3) The core measurement assembly is extracted from the top of the reactor to facilitate installation procedures and operational management.

IV. Drive Structure

The AC-600 uses two types of structures to drive the control-rod assemblies and compressed-water-rod assemblies. Magnetic jack drives are used for the 52 groups of control rods. The total height of the magnetic jack is 5,552 mm and the dimensions of the outside of the magnetic coil are 275 x 275 mm. The magnetic jack is composed of a position indicator, coil component, pressure-resistant casing component, a hook-claw component, and a drive-axle component. The pressure-resistant-vessel component is interconnected to the tube fitting on the drive mechanism on top of the pressure vessel. The drive axle passes through a connector which can be dismantled and connects to the control-rod assembly. When the coil portion on the outer surface of the pressure-resistant casing passes and shuts off the power according to a specified program, the magnetic field energy drives the armature to engage the hook claw into the cylindrical notch of the drive axle, which achieves step lifting, step lowering, or rod removal of the control-rod assembly. Its rated lifting load is 785 N and the maximum lifting force is 1,765.2 N. The lifting speed is 12 mm/second and the travel is 3.6 m.

The 85 groups of control rods are driven by hydraulic jacks. They are located in the spaces in the magnetic jacks at the top of the reactor. Their maximum outer diameter is 100 mm and they are 4,235 mm tall. They are composed of a pressure-resistant vessel component, support component, locking-latch component, drive-rod component, and connector, which can be dismantled. When it is necessary to lift the compressed-water rods from the reactor, the flow-rate regulation valves are opened, which creates a low-pressure region in the cavity in the upper part of the pressure-resistant casing. This causes a pressure differential to appear between it and the cavity within the reactor. This pressure differential pushes the compressed-water rods out of the core, and then closes the flow-rate regulation valves, equalizing the pressure in the upper and lower cavities of the pressure-resistant chamber, and the drive rod drops to the required position in the locking latch structure. When a compressed-water rod must be inserted into the core, the flow-rate regulation valve must again be opened and closed once. This aligns the drive rod and locking-latch structure in the trough and the compressed-water rod drops under its weight, enabling it to enter the core slowly.

Reactor Pressure-Vessel Design

906B0020F Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 10 No 5, Oct 89 pp 35-38 [MS received 25 May 89]

[Article by Zhang Jingcai [1728 2417 2088] of the Southwest Center for Reactor Engineering Research and Design, Chengdu: "Inquiry Into the AC-600 Pressure-Vessel Design"]

[Text] **Abstract**

This article offers a preliminary analysis of the design characteristics of the AC-600 reactor pressure vessel on the basis of the fundamental design principles for ALWR [advanced light-water reactor] and explores the primary technical measures required to guarantee the reliability of the AC-600 reactor pressure vessel.

Key words: advanced pressurized-water reactor, pressure vessel, safety, reliability.

I. Special Attributes of the AC-600 Pressure Vessel

There are no significant qualitative differences between the primary design parameters, functional requirements, load-carrying conditions, basic structure, and materials of China's AC-600 [reactor] pressure vessel (RPV) and a two-loop PWR-RPV. The main design characteristics concern these areas:

1. Total Application of All Laws, Regulations, and Standards

For present ALWR-RPV designs, a complete set of design laws and regulations or comprehensive technical regulations has not been compiled. Although these two items were previously included in the research plans,[1] they have not been dealt with, so designs at the present time must apply in full the complete set of design laws and regulations used for PWR-RPV (such as NRC laws and regulations and their guiding principles, ASME-Sec III and their appendices, ASTM standards applicable to PWR-RPV design, and so on) and make a full summarization and utilization of experiences in PWR-RPV design, manufacture, and operation over the past 30 years.

2. ALWR Design Lifespan

The design lifespan of an ALWR is 60 years. This characteristic means that the design lifespan of the AC-600's RPV is 20 years longer than the design lifespan of a PWR-RPV. The various problems brought on by the increased lifespan of the AC-600 RPV must receive conscientious consideration in RPV design, manufacture, and operation.

3. Fully Utilizing PWR-RPV Experience Over the Past 30 Years

About 230 PWR power stations are in operation at the present time and nearly 100 are under construction. The design, manufacture, and operation of these PWR-RPV's has provided extremely rich experience for the design, manufacture, and operation of the ALWR-RPV. Examples include guaranteeing sufficient durability and low-irradiation sensitivity of materials, the need for full adherence to laws and regulations in designs and conservative designing, ensuring high-quality welding and butt welding in manufacture, implementing effective and reliable non-destructive testing, improving the quality of operating personnel and reinforcing management over operations, and so on. Full application of these technical

and managerial experiences is an important guiding ideology in our design as well as one of the characteristics of the AC-600 RPV design.

II. Assuring Sufficient Broadness in AC-600 RPV Design

The safety design for the AC-600 must consider protection of the safety of the masses under accident conditions as well as the safety of the equipment design. For example, the design of the AC-600 RPV must guarantee its strong and enduring integrity and give it an obvious margin of safety to enable it to withstand all types of transient working conditions during its entire lifespan and guarantee its reliability. This means that the design probability for the destruction of the pressure vessel must be less than or equal to 1×10^{-7} pressure-vessel-years.

The most basic way to ensure that these design requirements are met is strict implementation of current PWR-RPV design standards, such as USNRC-10CFR50 appendices A, B, G, and H, ANSI18.2, ASME BPV-Sec III and XI and their appendices and related documents, and ASTM standards for RPV.

III. Preventing Brittle Fracturing

Preventing brittle fracturing of the RPV is always one of the most basic criteria to consider in RPV design. This question is even more important because of the longer lifespan of the AC-600 RPV. For this reason, adequate measures must be adopted to ensure the safety of the AC-600 RPV, particularly in preventing the occurrence of brittle fracturing of the RPV.

Brittle destruction is determined by three factors, the durability of materials (including irradiation), structural defects, and stress levels.

The United States proposed the first durability requirements for ferritic steel materials[2] in 1972. In the same year, an article[3] proposed clear limits for fracture durability and made them formal regulations in 1974 (NB-2300). They state that "consideration must be given to the effects of irradiation on the durability of materials in the cylindrical region of a reactor core and additional requirements should be set forth in specifications documents to ensure that the pressure vessel has sufficient fracture durability during its lifespan." They also added an Appendix G concerning "preventing extension fracturing." Appendix G considered the obvious defects inherent in a vessel in consideration of the principles of linear elastic fracture dynamics and further clarified the effects of defects. The USAEC 10CFR50 published Appendix G in 1973. This provided laws and regulations establishing baseline requirements for the durability of materials to prevent and control vessel fracturing, and it became one of the criteria for vessel design. It states that "materials should have sufficient fracture durability to prevent the occurrence of brittle fracturing at power stations in all types of operating patterns during water-pressure experiments and accident working conditions within the stipulated useful lifespan of the power station."[4] The design standards in Appendix A for 10CFR50 also made similar stipulations. The AC-600 RPV design must consider the adoption of these criteria.

Guaranteeing and improving the durability of materials is a basic way to prevent the occurrence of brittle fracture, but attention also must be given to degradations in the durability of materials due to fast-neutron irradiation during the lifespan of the vessel. The results of nearly 30 years of research on embrittlement of RPV materials by irradiation are stated concisely in NRC-RG1.99, 10CFR50 Appendix G, and ASME (Sec III and XI) and its appendices. These documents provide all types of safety measures for considering the effects of irradiation in current ALWR-RPV design and operation. They concern the durability of materials and its control, defect control, and load or stress-control technologies. Examples include the rise (ΔRT_{NDT} in reference temperature (RT_{NDT}), the platform energy limits and lower limits on the Charpy V-shaped-cut impact-test curve ($\Delta E/E$), irradiation monitoring (ASTME-185 and 10CFR50 Appendix H), in-service annealing (ASTME-509), in-service inspection (ASME Sec XI and appendices), operating restrictions (ASME Sec III Appendix G), various types of stress limits (ASME Sec III), and so on.

These standards were applied in the design of the AC-600 RPV to ensure sufficient fracture durability of the materials, take full account of the effects of irradiation (including aging and pre-strain effects), assuring an extremely low probability of the occurrence of fractures in the AC-600 RPV, and even preventing the possibility of brittle fractures occurring.

IV. Improving Core Design, Reducing Fast-Neutron Irradiation Fluence

To reduce irradiation embrittlement of the vessel material, the purity of the materials must be increased and the sensitivity to irradiation reduced, and hand design measures must be adopted in the physical design of the core, fuel management procedures, design of the internal structure of the reactor, and other areas to reduce fast-neutron irradiation fluence and keep the fast-neutron irradiation fluence in the vessel below 2×10^{19} neutrons/cm^2 during its 60-year lifespan. Consideration was given to the following:[5-7]

1. The physical design of the core should reduce the core power density, which means designing a low-power-density core, such as 70 to 75 kW/L, which can reduce the irradiation-fluence rate on the vessel.

2. A radial neutron reflecting layer is used in the internal structure of the reactor for effective reflection of fast neutrons to reduce the effects of fast-neutron irradiation on the vessel.

3. There should be an appropriate increase in the internal diameter of the RPV cylinder. If, for example, the inner diameter is increased from ϕ3352.8 mm (132 inches) to ϕ3987.8 mm (157 inches), the intermediate

distance between the core and the inner wall of the vessel would be increased, which would increase its capability of slowing fast neutrons and reduce the portion of fast neutrons which reach the inner wall of the vessel.

4. Improved fuel management in the core can reduce local radial peaks in fast-neutron fluence. Examples include placing fuel elements which have a partial burnup depth or even false elements in the core near the edges of the vessel wall.

5. Using a local neutron shield pad on the suspension basket within the reactor to reduce local neutron fluence peaks.

Adoption of all of these measures in the AC-600 RPV design can guarantee that the fast-neutron fluence during the 60-year lifespan of the AC-600 RPV will not exceed the fast-neutron fluence during the 40-year lifespan of existing PWR-RPV.

V. Assuring Optimum Performance of Materials

Materials are the foundation of RPV design and manufacture. To ensure that materials in the AC-600 RPV have sufficient durability, low sensitivity to irradiation, excellent welds, and superior internal quality and homogeneity, experience with PWR-RPV materials should be used as a foundation for reinforcing scientific research and improving materials technologies. The following measures were adopted for this purpose:

1. Pure materials should be selected as raw materials used to smelt the steel. It is best to use desulphurized pure iron for smelting the steel to ensure that residual contaminating elements (like Cu, P, S, As, Ti, Sb, B, V, and so on) are reduced to minimum levels.

2. Use an electric-arc furnace to smelt the steel, after which it should be put through a steel-encased precision smelting furnace to increase the purity and adjust the steel-water mixture to ensure that the optimum alloyed element mixture is attained.

3. Do the casting of ingots using vacuum degasification circulation and vacuum ingot-casting techniques to reduce the hydrogen and oxygen content of the steel ingots.

4. Casting flash grooves must be completely removed and the forging ratio should be increased during forging to ensure that internal defects are completely breeched and compacted.

5. Study and improve heat-processing technologies to ensure homogeneity in the performance of forged components and crystal fining.

Research on new technologies also should be undertaken, such as external forging technologies, three-point anvil-forging technologies, hollow (hollow-core) ingot-casting technologies, and so on, to make additional increases in the durability of materials, meet the need for larger cast components, and produce defect-free large cast components.

VI. Conclusion

This article examined questions related to the design of the AC-600 RPV, in particular the safety of the AC-600 RPV, and offered the author's views. Full summarization and application of experiences in designing, manufacturing, and operating PWR-RPV, a set of comprehensive conservative design standards for PWR-RPV, and study and solution of the various effects on the lifespan of ALWR-RPV, particularly the effects of irradiation embrittlement, can better ensure the safety and reliability requirements for ALWR-RPV.

These views are being proposed for exchange and discussion by colleagues to enable better AC-600 RPV design and preparatory work.

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Design of RCS, Major Auxiliary Systems

906B0020G Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 10 No 5, Oct 89 pp 39-44 [MS received 25 May 89]

[Article by Tan Zuo [6223 4373] of the Southwest Center for Reactor Engineering Research and Design, Chengdu: "Design Characteristics of AC-600 Reactor Coolant System and Main Auxiliary Systems"]

[Text] **Abstract**

This article introduces the design programs and characteristics of the AC-600 reactor coolant system (RCS), main equipment, and main auxiliary systems and compares them in terms of economy, safety, reliability, and other areas with existing 600 MW PWR's.

Key words: reactor coolant system, system simplification, economy.

I. Introduction

Work on the initial conceptual design for the AC-600 RCS and its auxiliary systems was completed in 1987 and was then revised and perfected. To meet economic, safety, and reliability requirements, the following guiding ideology was observed in the conceptual design of the AC-600 systems:

1. We tried as much as possible to use simple auxiliary equipment or did not place high demands on the auxiliary equipment, even to the point of not requiring any system and equipment-design programs for the auxiliary facilities and trying to simplify the systems as much as possible.

2. We took full advantage of mature and reliable experience in designing and building existing 600 MW PWR nuclear power plants to reduce scientific-research expenditures and basic investments.

3. We tried to use advanced and reliable technologies developed over the past few years to improve the systems and equipment operating characteristics to increase the utilization ratio of the power plant.

4. We completely separated normal operating systems and systems for implementing nuclear safety functions, and simplified system operations to improve power-plant operational safety and reliability.

II. Reactor Coolant System

The AC-600 RCS is composed of two loops connected in parallel with the reactor in a symmetrical configuration, a pressurizer, a pressure-relief tank and associated pipes, valves, and instruments. Each loop has a steam generator and two inverted shielded pumps welded to the bottom seal of the steam generator. The flow chart of the AC-600 RCS is illustrated in Figure 1. Table 1 lists the main parameters for the RCS.

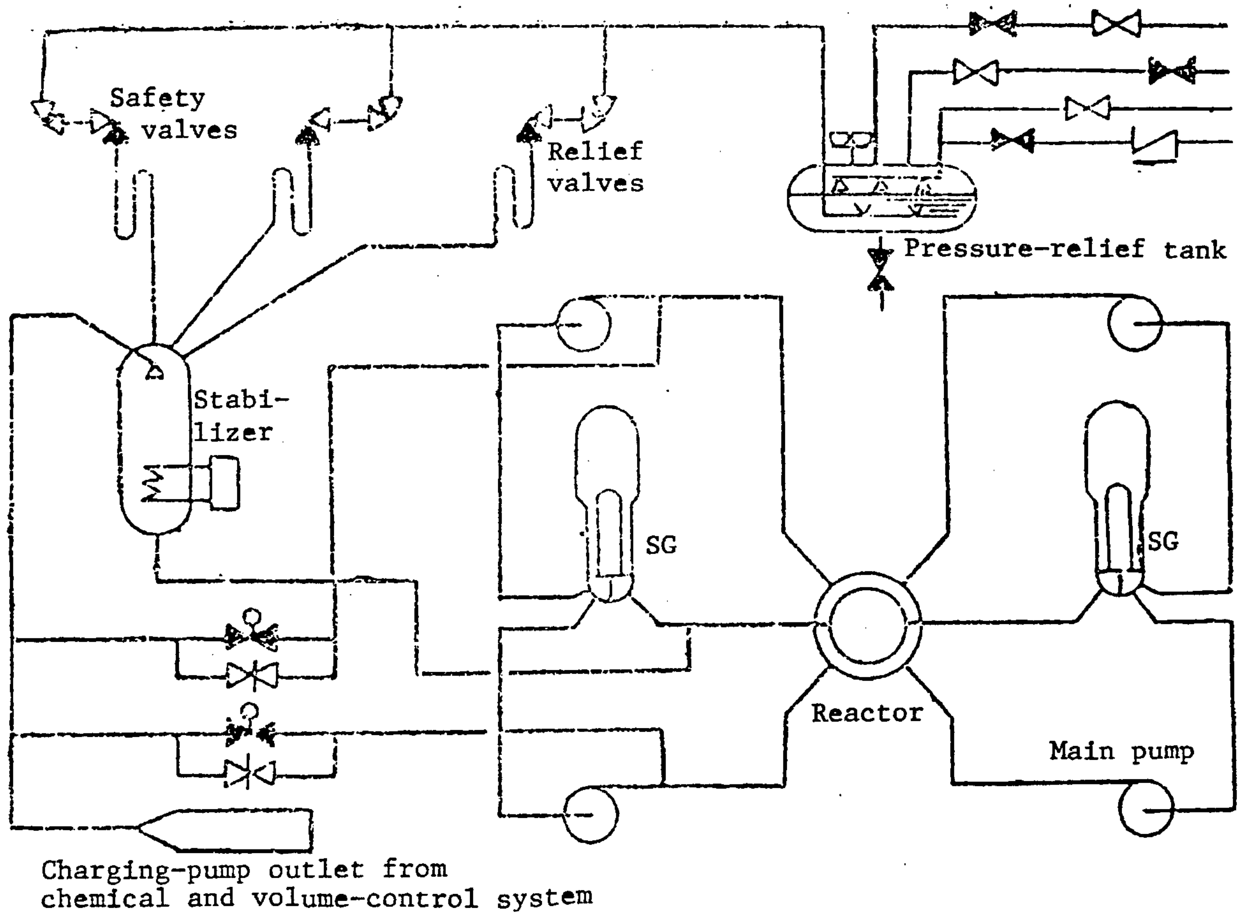

Figure 1. Simplified Flow Chart of RCS

Table 1. Primary Parameters for RCS

Reactor thermal power	1820 MW
Number of loops	2
Rated operating pressure	15.5 MPa
Design pressure	17.2 MPa
Design temperature	350°C
Reactor inlet temperature at rated power	287.7°C
Reactor outlet temperature at rated power	324.5°C
Coolant flow rate in each loop	4444.4 kg/s
Total volume of pressurizer	54.5 m^3
Dimensions of main pipe: int. diam. x wall thickness	
Hot leg	747 mm x 51 mm
Cold leg	531 mm x 40 mm

The configuration of the AC-600 RCS adhered to the principles of trying to improve the natural circulation capabilities of the system as much as possible to aid core flooding during a LOCA and to facilitate installation of equipment piping, in-service inspection and maintenance, and so on. A simplified diagram of the AC-600 RCS configuration is shown in Figure 2.

The AC-600 RCS design has the following characteristics compared to existing PWR power plants:

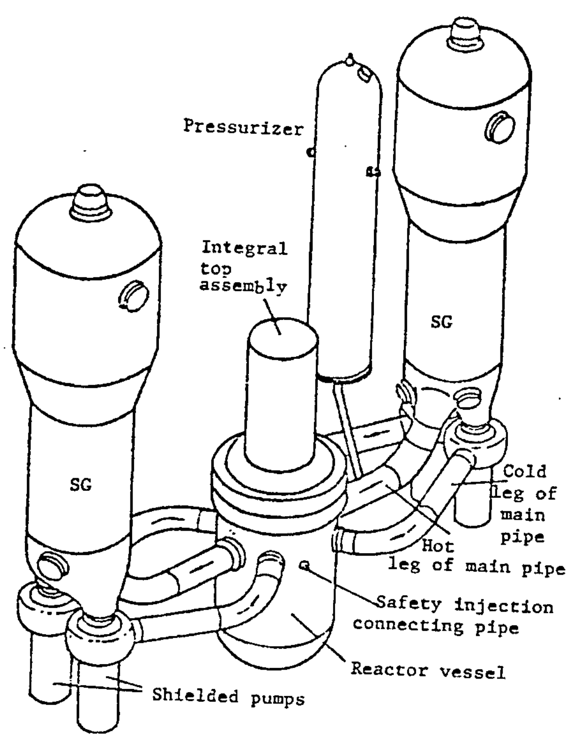

Figure 2. Diagram of Configuration of RCS

1. It is configured with two steam generators and four main pumps. The advantages are that the passive equipment steam generator still can use design and operating experience from existing power plants to maintain rather high thermal efficiency and rather low equipment expense. The smaller flow rate in the main pump of the active equipment enabled a switch to using a shielded pump. There is a reduction in the flow area in the cold leg of the main pipes, which lessens the risk of a large-break LOCA.

2. A mixed flow-type shielded pump is used as the main pump. Using a shielded pump in the system can eliminate axle seals, bearing lubricants, electric-motor cooling, and other auxiliary equipment serving the main pump. This simplifies system design, operation, and maintenance, and it eliminates the possibility of preventing normal operation of the main pump due to damage to the axle seals, LOCA's, and so on.

3. A larger-capacity pressurizer is used to provide more stability in RCS pressure during normal plant operation and to allow a slightly longer air-displacement time when a LOCA occurs, which benefits reactor safety.

4. Two main aspects of the RCS configuration are illustrated in Figure 2. One is that the main pump is welded directly to the outlet side of the steam generator, which eliminates the intermediate U-shaped segment between the steam generator and main pump in existing PWR power plants. This reduces system fluid resistance, eliminates complicated main-pump supports, reduces design analysis and equipment expenditures, and helps maintenance and in-service inspection. This configuration also means that no point in the main pipes is lower than the pressure casing flange, which reduces the probability of core exposure in LOCA situations. The second main aspect is the use of a curved pipe with a large radius of curvature from the main pump outlet to the inlet segment of the main pipe at the reactor. This reduces the number of welds in the curved pipe, improves system operation reliability, reduces fluid resistance in the system, and increases the system's natural-circulation capabilities.

III. RCS Equipment

1. Reactor Coolant Pumps

One of the main differences between the AC-600 and existing PWR power plants is that the former uses shielded pumps as reactor coolant pumps. Their main design parameters are shown in Table 2.

Table 2. Primary Design Parameters of Reactor Coolant Pumps

Type	**Vertical, single-absorption, single-stage mixed-flow shielded pump**
Design flow rate	2222 kg/s
Design lift	71 mH_2O
Cold-state start-up power	2600 kW

Table 2. Primary Design Parameters of Reactor Coolant Pumps (Continued)

Type	Vertical, single-absorption, single-stage mixed-flow shielded pump
Design net positive upward absorption height	49 mH_2O
Minimum start-up pressure	2.45 MPa
Power-supply voltage	6000 V
Pump rotation speed	1450 rpm
Total height of generator	6 m
Dry weight of pump casing	$18x10^3$ kg

The main pumps are connected directly to the electric motors, with the electric motors above and the pumps below. The pump cavities are linked to the electric-motor cavities. The pump intakes are welded directly to the outlet cavities of the lower seals of the steam generators. The pump cavities and electric-motor cavities are linked to the main pipes to create a reactor-coolant pressure margin.

The shielded pumps in the AC-600 have these advantages compared to mechanical shaft-seal pumps used in existing PWR power plants:

(1) The main pumps are installed upside-down on the lower seals of the steam generators and the electric motors are placed below the pumps. The gas in the medium inside the electric-motor cavities can enter the system automatically and cannot accumulate in the electric-motor bearings, which aids bearing lubrication and reliable operation.

(2) Placing the electric motors in the lowest position can prevent heat transfer to the electric motors via natural convection through cracks in the heat shields, preventing a rise in temperature in the electric motors.

(3) The rotors of the shielded pumps have excellent stability, and no mechanical damage occurs in the rotors and machinery when there is short-term excessive speed in single-pump operation.

(4) Certain measures were adopted in the design to forestall damage to the electric motors if there is a cutoff in equipment coolant water for a certain period of time. When the water jackets of the shielded electric motors are full of water, the equipment cooling water can be cut off for 10 minutes. It can be cut off for 5 minutes during water-jacket air displacement.

One of the main problems involved in replacing the mechanical shaft-seal pumps in existing PWR power plants with shielded pumps is the random rotation flow rate after a power cutoff. Using an electrical and mechanical arrangement can increase the rotational inertia of the main pumps and lengthen the radial flow-rate time after a power cutoff to the main pumps. Preliminary analysis indicates that the arrangement used in the AC-600's main pumps to increase the weight of the impellers in the main pumps can satisfy the reactor's requirement of random rotation in a power cutoff to the main pumps.

2. Steam Generators

The main concern in the design of the AC-600's steam generators is to increase the reliability of equipment operation. The main design parameters are shown in Table 3 and a simplified structural diagram is illustrated in Figure 3.

Table 3. Primary Design Parameters of Steam Generators

Type	F-type inverted U-tube, vertical
Heat-transfer-tube material	Incoloy-800
Heat-transfer-tube dimensions (outer diam. x wall thickness)	17 mm x 1.0 mm
Heat-transfer area	5109.7 m^2
Steam outlet pressure at rated power	5.88 MPa
Steam outlet temperature at rated power	274.3°C
Steam output at rated power	501 kg/s
Feedwater temperature	225°C
Total height	about or similar to 20 m

The AC-600's steam generators have the following advantages compared to existing PWR power plants:

(1) To deal with corrosion damage to the heat-transfer pipes due to accumulation of sediments on the pipe plates during steam-generator operation, the design gave special emphasis to ways to clean the pipe plates.

(2) To effectively eliminate cracks between the heat-transfer pipes and pipe plates, full-depth hydraulic pipe expansion was used.

(3) To enable closer placement of the heat-transfer pipe bundles and increase the heat-transfer coefficient of the pipe walls, heat-transfer pipe bundles with smaller-diameter pipes were used.

(4) To reduce the contact between the heat-transfer pipes and their supporting plates and wear due to the resulting vibration, the normal circular holes in the support plates were changed to cloverleaf-shaped holes.

(5) The J-shaped spray connecting pipes were welded on top of the feedwater loop pipes to prevent water hammer phenomena in regular feedwater loop pipes.

IV. Reactor Coolant Auxiliary Systems

The biggest difference between the AC-600's reactor coolant auxiliary systems and existing PWR power plants is a substantial simplification of these systems and even the elimination of some systems. The principles for simplification of these systems and elimination of some systems involve the following areas:

1. Because a passive safety system is used, several auxiliary systems in existing PWR power plants which support the operation of matching safety systems are simplified or even eliminated. Examples include part of

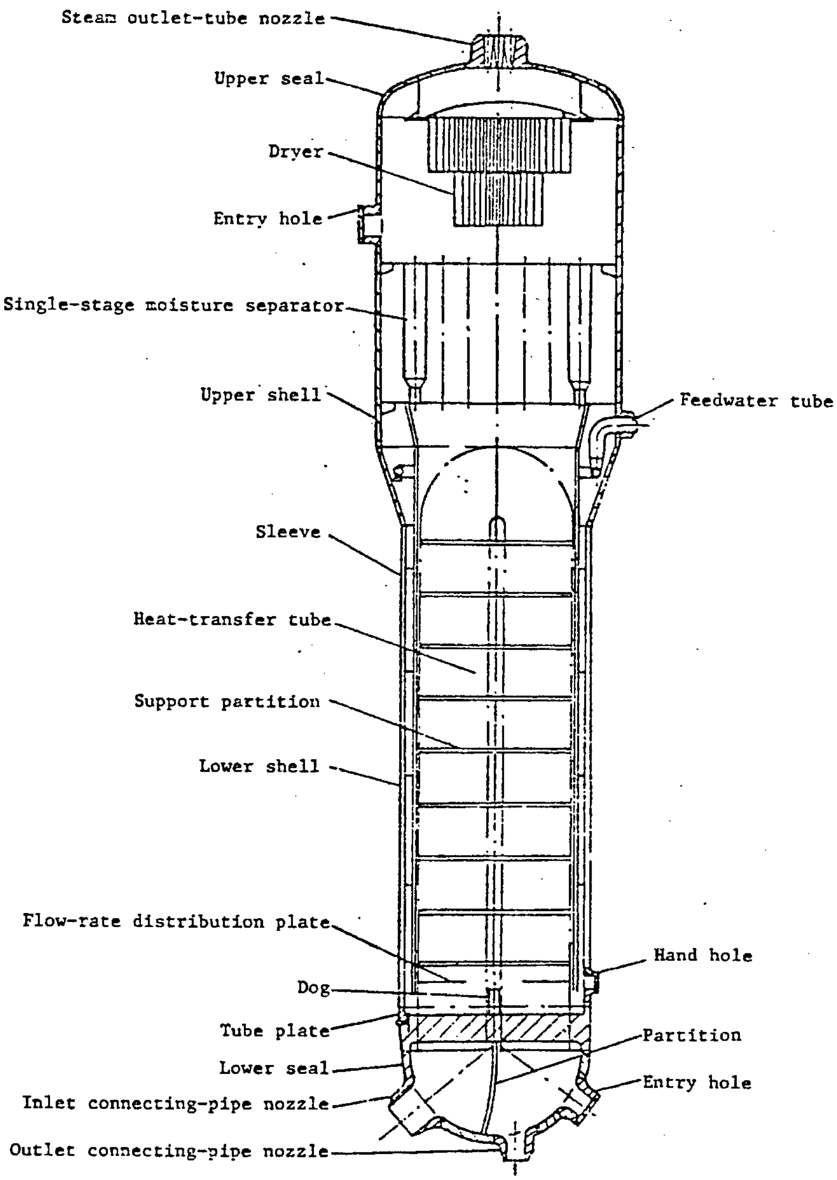

Figure 3. Structure of Steam Generator

the auxiliary feedwater system and RHR system which are an important part of the working process safety system as well as the chemical and volume-control system which is a part of the nuclear-safety functions, the boron recovery system, and so on, all of which have been eliminated or simplified. All their nuclear-safety functions are performed by passive safety systems. The non-nuclear-safety functions of the auxiliary feedwater system and RHR system are performed by startup and shutdown of the reactor feedwater system and an improved depleted-fuel-tank cooling system.

2. Improvements in the core design make the reactor less dependent on reactor coolant auxiliary systems during operation, thereby simplifying these auxiliary systems. Examples include using control rods and gray rods for power plant load-following, reducing the critical boron concentration in reactor operation, lengthening the fuel cycle time, and so on, all of which reduces reliance on soluble boron regulation during operation and enables the AC-600 to eliminate the boron heat following and boron recovery systems in existing PWR power plants.

3. There is reduced reliance on or even basically no need for auxiliary system equipment in the RCS, and reactor coolant auxiliary systems are simplified. Using shielded pumps in the AC-600 as the main pumps, for example,

permits elimination of main-pump shaft seal-water systems which serve the mechanically sealed main pumps in existing standard PWR power plants and reduces the upward filling and downward leakage capacity of the chemical and volume-control system. Another example is the addition of greater volume in the pressurizer for the AC-600 compared to that in existing PWR power plants, which increases the range of fluctuation in pressure and temperature absorption by the pressurizer itself and reduces the involvement of the chemical and volume-control system.

In the area of hardware for the chemical and volume-control system, main-pump shaft seal-water systems and surplus drainage channels which are complex and have rather high safety requirements in existing PWR power plants have been eliminated, which conserves investments.

In another area, the chemical and volume-control system is no longer responsible for any functions related to reactor safety, so this system does not have to satisfy single-breakdown criteria. The requirements for emergency power sources, coolant water sources, and other safety grades have enabled simplification of system design, analysis, configuration, and operation. Moreover, a reduction in the critical boron concentration for core operation means that the highest boron concentration in the working medium for the chemical and volume-control system was reduced from 7,000 ppm (the boron concentration required for boron injection to meet emergency reactor shutdown requirements was 12,000 ppm) to 4,000 ppm or even lower. The crystallization temperature of boron at this concentration is below 0°C, so with the exception of conventional measures to prevent freezing in power plants, no additional heat tracking facilities are required for containers and pipelines filled with boron.

V. Conclusion

Economics and safety are the two obstructions to development of nuclear power in the world. Making obvious breakthroughs in the economy and safety of future nuclear power plants is the key to the recovery of the nuclear power industry. In practice in the past, however, it was generally felt that economy and safety were mutually restrictive, meaning that making nuclear power plants more economical required reductions in safety requirements, or that increasing power-plant safety required paying an additional economic cost. This put nuclear power development in a bind which it could not break out of.

During the process of APWR power-plant R&D, we made system simplification one of the primary means for breaking out of this bind. The preliminary conceptual design and analysis for the AC-600 RCS and auxiliary systems showed that rational simplification of these systems could reduce the basic costs, maintenance, inspection, and other operating costs of power plants and improve the economy of power plants, and that further system simplification and ease of operation could decrease the dependence of reactor operation on the operation of auxiliary systems and increase the reliability and safety of power plant operation.

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Plant Layout

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[Article by Xu Changrong [1776 7022 2837] and He Weidong [0735 5898 2639] of the Southwest Center for Reactor Engineering Research and Design, Chengdu: "AC-600 Plant Building Layout"]

[Text] **Abstract**

This article provides a brief explanation and discussion of the layout of the AC-600 nuclear power plant reactor building and nuclear auxiliary building.

Key words: passive advanced pressurized-water reactor, overall layout of nuclear power plant.

On the basis of the conceptual design for the AC-600 and in reference to the overall planar configuration of the Westinghouse Corporation's AP-600 nuclear power plant in the United States, we have prepared a preliminary layout for the AC-600 nuclear-power-plant reactor building, given preliminary consideration to the layout of the nuclear auxiliary building and fuel building, and discussed several questions related to the plant building configuration.

I. Principles of the Layout

The overall planar configuration of the AC-600 nuclear power plant and the plant layout were prepared in accordance with these principles:

1. Simplification and compaction of the overall planar configuration to reduce the plant building area.

2. Placing radioactive plant buildings close to the reactor building layout and clearly demarcating "clean" areas and "possibly contaminated" areas to reduce irradiation doses for personnel.

3. Using a dual-layer cylindrical containment vessel to increase safety and reliability.

4. Configuring all the passive special-purpose safety facilities within the reactor building to reduce the earthquake-resistant plant building area.

5. Adopting identical passive safety concepts for the configuration of the reactor building to improve safety.

6. Comprehensive consideration of constructability, operatability, maintainability, and construction costs.

II. Reactor Building Layout

The containment vessel and its internal structures form the reactor building. The reactor, RCS, and passive special-purpose safety facilities are located in this building. Figure 1 shows vertical and horizontal cross sections of the reactor building.

The dual-layer cylindrical containment vessel includes two parts: an internal steel containment vessel and an external concrete containment vessel. Because the steel containment vessel is the main pressure-bearing component and thus prevents the leakage of large amounts of radiation during an accident, shock to the external components is unavoidable, so a solid thick-walled concrete containment vessel must be installed. Moreover, the concrete containment vessel also plays a role in biological shielding. The circular space between the two-layered containment vessel, the space at the top, and the air outlet at the bottom of the containment vessel form the passive containment-vessel cooling system. The air enters the circular space from the inlet at the bottom and relies on natural circulation to rise by convection to the outlet at the top where it is vented into the atmosphere, cooling the containment vessel and discharging the small amount of radioactive steam leaked from the steel containment vessel into the atmosphere after filtering. The steel containment vessel is about 35 meters in diameter and about 70 meters tall.

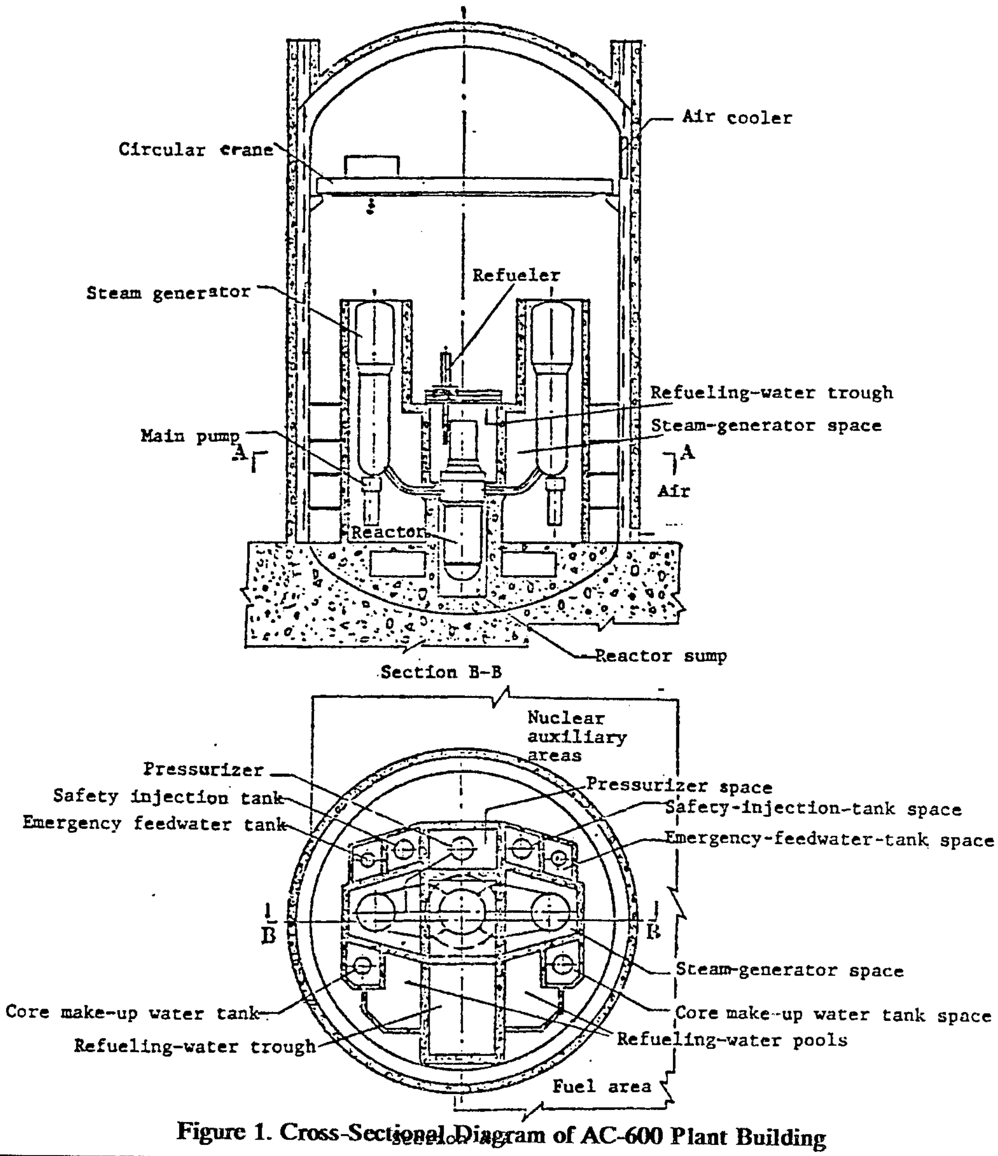

Figure 1. Cross-Sectional Diagram of AC-600 Plant Building

The RCS is a symmetrical, simplified, "four-inlet/two-outlet" two-loop system centered on the reactor. Each loop includes a steam generator, two shielded pumps installed in an inverted fashion on the seal at the bottom of the steam generators, one hot-leg pipe, and two cold-leg pipes. The pressurizer is linked to one of the hot-leg pipes. The reactor is located in the middle of the lower part of the containment vessel with its centerline offset by about 1 meter from the centerline of the containment vessel. The two steam generators are located about 10 meters from the centerline of the reactor. A pressurizer, two emergency feedwater tanks, and two safety injection tanks are installed at one side of the steam generator—reactor—steam-generator centerline. Two core make-up water tanks and two refueling pools with a volume of 850 m^3 each are located on the other side. A refueling water trough about 7.5 meters wide, about 20 meters long, and about 9.5 meters deep is located on top of the reactor. In accordance with the design requirement of "physical isolation," the reactor, steam generators, pressurizer, and water tanks (pools) are installed in their own concrete isolation chambers. Sufficient space has been left between the equipment and isolation chambers to facilitate installation, inspection, and repair. To increase the height differential of the fluid and the primary loop and thereby increase the drive force, the water-tank datum mark was raised as much as possible.

The operating console datum mark is 23 meters. The refueling water pool is located below the operating console.

A circular crane is installed on top of the containment vessel. The crane datum mark is mainly determined by the height of the steam-generator hoists, whereas the height of the containment vessel is mainly determined by the height of the crane. The containment vessel, a "colossus," obviously is not lenient toward hoisting, storing, inspecting, and repairing the large primary equipment.

III. Auxiliary Nuclear Building and Fuel Building Layout

The primary layout within the nuclear auxiliary building for the AC-600 nuclear power plant includes the passive RHR system, chemical and volume-control system, equipment cooling-water system, auxiliary feedwater system, industrial waste water, waste gas, residue waste-processing system, boron recovery system, and so on. Excluding the special-purpose safety systems, the other parts of this plant building are similar to those in the nuclear auxiliary building at existing PWR nuclear power plants, so there is no obvious reduction in the area of the plant building.

Because the technical design, equipment configuration, and other aspects of the AC-600 nuclear power plant's fuel removal and storage systems are basically the same as those in the fuel operations building of existing nuclear power plants, the plant buildings for both types basically cover the same area.

The reactor building, nuclear auxiliary building, and fuel building are located on the same foundation plate.

IV. Issues and Discussion

Several questions related to the layout of the AC-600 nuclear power plant building should be explained and discussed.

1. There is no overall reduction in the plant building area of the AC-600 nuclear power plant compared to existing 600 MW_e nuclear power plants. The 2.9-meter increase in the diameter of the steel containment vessel increases the area of the reactor building by about 19 percent. There is only a small reduction in the area of the nuclear auxiliary building and no basic change in the area of the fuel building. For this reason, there is no significant change in the total area of the three plant buildings. The most recent information for the AP-600 indicates that the area of the AP-600 reactor building also was increased by 11 percent, but the earthquake-resistant plant building area was decreased by three-fourths and the total area by 51 percent. The reduction in the plant building area, particularly in the earthquake-resistant plant building area, can directly reduce construction costs and shorten construction schedules. Thus, as the AC-600 system designs are intensified and in reference to the AP-600 plant building layout, a substantial reduction in the plant building area is very important.

2. The refueling water pool is the supplementary feedwater source for the low-pressure safety injection system and the refueling water trough inside the containment vessel. Because low-pressure safety injection pumps were retained for the AC-600 low-pressure safety injection system, the refueling water pool is also located outside the containment vessel. This helps reduce the size of the containment vessel and aids in fuel water pool cleaning and connecting pipes in the refueling-water-pool cooling system.

3. The passive containment-vessel cooling system plays a key role in directly transferring heat from the containment vessel into the air. The containment-vessel structural design needs further exploration and experimental testing and verification. Continual improvements are also being made in the AP-600 containment-vessel design and simulation experiments are now in progress to guarantee that it can perform the above function.

4. Although modularization technologies are common, using them widely in nuclear power plants requires further study of such things as rational demarcation of modules, module installation, and so on.

5. Both in the AC-600 and AP-600, the use of a cylindrical containment vessel creates problems for

the reactor building layout and increases the diameter of the containment vessel. If a spherical containment vessel is used, the reactor building layout is easier, and some of the nuclear auxiliary systems and equipment can be located inside the containment vessel. The actual choice of the type of containment vessel requires further debate on programs.

In summary, because the debate on the AC-600 program is in the preliminary stages, it is hard to provide greater detail on the plant building layout. In view of the problems mentioned above, we should make full use of the technical and economic advantages of the AC-600 and arrange good total area configurations and the nuclear power plant building layout.

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